Westinghouse Non-Proprietary Class 3

WCAP-16307-NP

September 2004

# Beaver Valley Power Station Extended Power Uprate Licensing Report Supplemental Information

Westinghouse

**WCAP-16307-NP**

# Beaver Valley Power Station Extended Power Uprate Licensing Report Supplemental Information

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**September 2004**

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## LIST OF ACRONYMS

| | |
|---|---|
| ANC | Advanced Nodal Code |
| ASME | American Society of Mechanical Engineers |
| AVB | Anti-Vibration Bar |
| BVPS | Beaver Valley Power Station |
| CFR | Code of Federal Regulations |
| CLHs | Capped Latch Housings |
| COMS | Cold Overpressure Mitigation System |
| CRDM | Control Rod Drive Mechanism |
| CUF | Cumulative Usage Factor |
| DFBN | Debris Filter Bottom Nozzle |
| DNB | Departure from Nucleate Boiling |
| ECT | Eddy Current Testing |
| FENOC | FirstEnergy Nuclear Operating Company |
| EFPD | Effective Full Power Days |
| EFPY | Effective Full Power Years |
| EPU | Extended Power Uprate |
| ERG | Emergency Response Guidelines |
| ESFAS | Engineered Safety Feature Actuation System |
| FIV | Flow-induced Vibration |
| FSRF | Fatigue Strength Reduction Factor |
| ID | Inside Diameter |
| IFM | Intermediate Flow Mixing |
| IRI | Incomplete Rod Insertion |
| KSI | Pipe Stress |
| LBB | Leak Before Break |
| LOCA | Loss of Coolant Accident |
| LPD | Low Pressure Drop |
| LSIV | Loop Stop Isolation Valve |
| LWS | Laser Welded Sleeve |
| MCO | Moisture Carryover |
| MIFM | Modified Intermediate Flow Mixer |
| MLPD | Modified Low-Pressure Drop |
| MWt | Megawatts Thermal |
| NRC | Nuclear Regulatory Commission |
| NSSS | Nuclear Steam Supply Systems |
| OD | Outside Diameter |
| ODSCC | Outside Diameter Stress Corrosion Cracking |
| OSG | Original Steam Generator |
| PCWG | Power Capability Working Group |
| PDA | Percent Degraded Area |
| PIP | Plug In Plug |
| PWSCC | Primary Water Stress Corrosion Cracking |
| RCS | Reactor Coolant System |

## LIST OF ACRONYMS (cont.)

| | |
|---|---|
| RFA | Robust Fuel Assembly |
| RPC | Rotating Probe Coil |
| RRB | Reduced Rod Bow |
| RSG | Replacement Steam Generator |
| RSR | Relative Stability Ratio |
| RTN | Removable Top Nozzle |
| RTP | Rated Thermal Power |
| RTS | Reactor Trip System |
| SGTP | Steam Generator Tube Plugging |
| SGTR | Steam Generator Tube Rupture |
| SLB | Steamline Break |
| STD | Standard |
| TDF | Thermal Design Flow |
| TSP | Tube Support Plate |
| UFSAR | Updated Final Safety Analysis Report |

## ABSTRACT

This report provides information of a proprietary nature in support of the License Amendment Request for Beaver Valley Power Station for the Extended Power Uprate. Included in this report are portions of Section 4.0 "NSSS Components." Section 5.1 "Initial Condition Uncertainties" and portions of Section 6.0, "Fuel Analysis" that contain proprietary information.

# 4 NSSS COMPONENTS

The EPU Project included analyses and evaluations for the NSSS components at EPU conditions. The primary EPU-related inputs to these analyses and evaluations are the NSSS design (PCWG) parameters (Section 2.1.1), the NSSS design transients (Section 2.2.1), and the auxiliary equipment design transients (Section 2.2.2). In addition to these primary inputs, the NSSS component evaluations used the existing component design basis information to assess the impact of EPU. The NSSS component evaluations were performed to confirm that the NSSS components continue to comply with applicable licensing requirements and industry codes at EPU conditions.

The following NSSS components are addressed in this section:

- Reactor Vessel
- Reactor Pressure Vessel System (i.e., Reactor Internals)
- Fuel Assemblies
- Control Rod Drive Mechanisms and Capped Latch Housings
- Reactor Coolant Loop Piping and Supports*
- Reactor Coolant Pumps
- Steam Generators
- Pressurizer
- NSSS Auxiliary Equipment
- Loop Stop Isolation Valves

The analyses and evaluations presented in this section support operation of BVPS-1 at EPU conditions with the Model 54F replacement steam generators (RSGs) and BVPS-2 at EPU conditions with the original steam generators. The analyses and evaluations for EPU conditions bound and support operation at the current power level, which supports the staged implementation of EPU at BVPS-1 and BVPS-2.

## 4.1 REACTOR VESSEL

The EPU Project included analyses and evaluations for the reactor vessel, including structural integrity in accordance with the requirements of the ASME Code and integrity in accordance with NRC requirements for radiation embrittlement and pressurized thermal shock. These analyses and evaluations are addressed in this section.

### 4.1.1 Reactor Vessel Structural Integrity

#### 4.1.1.1 Introduction

Evaluations were performed for the various regions of the BVPS-1 and BVPS-2 reactor vessels to determine the stress and fatigue usage effects of NSSS operation at the revised operating conditions of the EPU Project throughout the current plant operating license. The revised operating parameters identify

* Pressurizer Surge Line Stratification and Application of Leak-Before-Break (LBB) Methodology are addressed in this section. The reactor coolant loop piping and supports are addressed in Section 8.3.

vessel inlet and vessel outlet temperatures that define the steady state operating temperatures for the reactor vessel for a range of high and low temperature operation. The design transients for the reactor coolant system (RCS) define the temperature and pressure responses for a variety of transients that can occur during either high temperature or low temperature operation. In addition, new LOCA and seismic interface loads were defined for EPU. The revised transient temperature and pressure variations, along with seismic and LOCA loads, may affect both the maximum ranges of primary plus secondary stress intensity and the maximum cumulative fatigue usage factors for the reactor vessel. The evaluations assess the effects on the maximum ranges of stress intensity and fatigue usage factors at the most limiting locations in each of the regions of the reactor vessel as identified in the reactor vessel stress report and addenda.

#### 4.1.1.2 Input Parameters and Assumptions

The key input parameters include the PCWG parameters in Section 2.1.1, the NSSS component design transients in Section 2.2.1, and the reactor vessel/reactor internals interface loads. The temperatures and pressures considered in the reactor vessel structural evaluation for the PCWG cases are as follows:

| | |
|---|---|
| Normal Operating Pressure: | 2250 psig (conservative with respect to PCWG parameters) |
| Normal Operating Temperatures | |
| Vessel Inlet Temperature Range: | 528.5° to 543.1°F |
| Vessel Outlet Temperature Range: | 603.9° to 617.0°F |
| Zero Load Temperature: | 547°F |

The reactor vessel was previously evaluated as part of the increased steam generator tube plugging (SGTP) and 1.4% uprating projects. The structural evaluations that were performed are included in calculations and addenda to the reactor vessel stress report.

The reactor vessel outlet nozzle internal surfaces are in contact with vessel outlet water during normal reactor operation. The closure head internal surfaces including those of the CRDM housings are in contact with water at the head temperature ($T_{head}$) which is near the vessel outlet temperature. The remainder of the reactor vessel internal surfaces are in contact with vessel inlet water during normal reactor operation. Therefore, the vessel outlet temperature and $T_{hot}$ transient temperature variations apply for the outlet nozzles. The head temperature and the $T_{hot}$ transient temperature variations apply for the vessel closure and CRDM housings. The vessel inlet temperature and $T_{cold}$ transient temperature variations apply for the remainder of the reactor vessel.

In order that the most conservative results from previous analyses are maintained, the evaluations also assume that the plant may operate at the EPU high or low temperature operating conditions or in accordance with the original design basis for the entire term of the operating license.

#### 4.1.1.3 Description of Analyses and Evaluations

The revised NSSS design transients for the EPU Project were reviewed and compared to the original design basis transients. This transient review determined which revised transients are more severe than their design basis counterparts by comparing rates, magnitudes and durations of the transient temperature variations as well as the magnitudes of the pressure variations. Based on this review, a determination of which revised transients must be considered in the stress evaluations was made.

The transient review concluded that six of the $T_{hot}$ transient temperature variations were more severe for the EPU than for the original design basis. The more severe $T_{hot}$ design transients include:

1. Plant (Unit) Loading at 5% of Full Power per Minute
2. Plant (Unit) Unloading at 5% of Full Power per Minute
3. Step Load Increase of 10% of Full Power
4. Step Load Decrease of 10% of Full Power
5. Loss of Load
6. Loss of Power

In addition, RCS Cold Overpressurization was included as a new design transient for BVPS-1 and Refueling and Inadvertent Safety Injection were included as new design transients for BVPS-2. The stress intensities for these revised $T_{hot}$ transients and new transients were examined to determine their effect on the maximum ranges of stress intensities for the outlet nozzle, main closure flange assembly and CRDM housings. Also for BVPS-1, the analysis in the original stress report was performed with the assumption that the upper head temperature was at $T_{cold}$ temperature during operation but it has been determined that the head temperature is closer to $T_{hot}$. Thus, for BVPS-1 the $T_{hot}$ head temperature was addressed in the Main Closure Flange and CRDM Housing evaluation in addition to the influences of the $T_{hot}$ transients.

The transient review also concluded that six of the $T_{cold}$ transient temperature variations were more severe than the original design basis. The more severe $T_{cold}$ design transients include:

1. Plant (Unit) Loading at 5% of Full Power per Minute
2. Plant (Unit) Unloading at 5% of Full Power per Minute
3. Large Step Load Decrease
4. Loss of Load
5. Loss of Power
6. Reactor Trip from Full Power

RCS Cold Overpressurization was included as a new design transient for BVPS-1 and Refueling and Inadvertent Safety Injection were included as new design transients for BVPS-2. These revised $T_{cold}$ transients and new transients were applied to all regions of the reactor vessel pressure boundary (other than the main closure, CRDM housings and outlet nozzles), including the vessel inlet nozzles, bottom head, vessel shell, and core support guides that are in contact with vessel inlet water ($T_{cold}$) during normal operation. The change in the transient thermal stress due to the revised and new transient temperature variations were calculated. The incremental thermal stress changes were then factored into the previous stress intensities reported and the effects of the changes on the maximum ranges of stress intensity were evaluated.

Changes in pressure variation were also noted for the EPU transients. There are five pressure transients identified as being more severe for the EPU than for the original design basis:

1. Large Step Load Decrease
2. Loss of Load
3. Loss of Power
4. Loss of Flow
5. Reactor Trip from Full Power

The incremental pressure stresses were calculated by scaling the original design transient pressure stresses proportional to the changes in the pressure variation, since pressure stress is directly proportional to pressure. The changes in pressure stresses were also added to the revised stress intensities for both the $T_{hot}$ and $T_{cold}$ regions where appropriate. For BVPS-1, the RCS Cold Overpressurization transient was also evaluated for pressure stress contributions using the scaling method. For BVPS-2, the Inadvertent Safety Injection and RCS Cold Overpressurization transients were also evaluated for pressure stress contributions using the scaling method. These revised and new pressure transients were considered in the evaluations of the various vessel regions even if the applicable transient temperature variation was unaffected.

Where applicable, the maximum and minimum stress intensity ranges and fatigue usage factors were revised to reflect the changes in the transients due to the EPU for various regions. In other cases, the original design basis stress analysis remains conservative, and new calculations were not necessary. Therefore, the maximum stress intensity ranges and fatigue usage factors reported in the original reactor vessel stress report and addenda continue to govern.

Revised seismic and LOCA loads for the reactor vessel interfaces with the reactor internals at the main closure flange ledges, the outlet nozzle internal projections and the core support pads were calculated. These seismic and LOCA loads were compared to the previous faulted condition loads.

For BVPS-1, it was found that only the Steam Pipe Break transient was covered in these evaluations and as a result, the main closure flange, outlet nozzle and core support guides needed to be evaluated for faulted conditions. Evaluations had previously been performed for a similar 3-loop plant with geometry and materials identical to the BVPS-1 vessel. The faulted conditions used in these evaluations bound the current faulted conditions for the EPU Project. Therefore, the acceptable conclusions from these evaluations are applicable to BVPS-1.

For BVPS-2, it was found that all of the seismic and LOCA loads at the reactor vessel/reactor internals interfaces exceeded the loads that were previously considered in the reactor vessel stress report. Therefore, calculations were performed to calculate the design, upset and faulted condition stress intensities. Stress intensities were calculated for the limiting locations at the interfaces and were evaluated, as required. The maximum stress intensity results were then compared to the appropriate ASME Section III acceptance criteria.

No computer codes were used for any of the reactor vessel structural evaluations performed for the EPU Project.

### 4.1.1.4 Acceptance Criteria and Results

The acceptance criteria for the reactor vessel structural analyses and evaluations for BVPS-1 are in accordance with the applicable requirements of the 1968 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through Summer 1970 (Reference 1) and the 1974 Edition (Reference 2) for faulted conditions. The acceptance criteria for BVPS-2 are in accordance with the applicable requirements of the 1971 Edition of Section III of the ASME Boiler and Pressure Vessel Code with Addenda through Summer 1972 (Reference 3) and the 1974 Edition (Reference 2) for faulted conditions. The applicable acceptance criteria are as follows:

- The maximum range of primary plus secondary stress intensity resulting from normal and upset condition design transient mechanical and thermal loads shall not exceed $3S_m$ at operating temperature, in accordance with Paragraph N-414.4 of Reference 1 for BVPS-1 and with Paragraph NB-3222.2 of Reference 3 for BVPS-2.

- The maximum cumulative usage factor resulting from the peak stress intensities due to normal and upset condition design mechanical and thermal loads shall not exceed 1.0, in accordance with the procedure outlined in Paragraph NB-415.2 of Reference 1 for BVPS-1 and with Paragraph NB-3222.4 of Reference 3 for BVPS-2.

- The faulted conditions shall meet the component criteria of Appendix F of the ASME Boiler and Pressure Vessel Code, Section III, 1974 Edition (Reference 2: The 1974 Edition was used since Appendix F was not yet included in the 1968 and 1971 Editions).

  For BVPS-1 faulted condition analyses of the main closure flange assembly, outlet nozzles and core support guides, the general primary membrane stress intensity limits are $2.4S_m$ for the closure studs and core support guides, and $0.7S_U$ for the closure head and vessel flanges, outlet nozzle and vessel shell. The primary membrane plus bending limits are $3.6S_m$ for the closure studs, and $1.05S_U$ for the closure head, vessel flange, outlet nozzle and vessel shell.

  For BVPS-2 faulted condition analyses of the main closure flange assembly and outlet nozzles, the general primary membrane stress intensities in the ASME SA-508, Class 2 material for the flange and nozzle forgings shall not exceed $0.7S_U$. The primary membrane plus bending stress intensities for the flanges and nozzles shall not exceed $1.05S_U$. The maximum stress in the closure studs during the faulted conditions shall not exceed $3.6S_m$ for SA-540 bolting material. For the core support guide faulted condition analysis, the general primary membrane stress intensity limits are $2.4S_m$ for the Alloy 600 pads and $0.7S_U$ for the low-alloy steel vessel shell. The primary membrane plus bending limits are $3.6S_m$ for the pads and $1.05S_U$ for the vessel shell.

The EPU affects several of the maximum ranges of primary plus secondary stress intensity reported in the BVPS-1 and BVPS-2 reactor vessel stress reports.

For BVPS-1, the maximum range increases for the outlet nozzle and core support guides but remain within the applicable ASME Boiler and Pressure Vessel Code, Section III limit. All of the fatigue usage factors were changed but remained under the acceptable limit of 1.0. The highest usage factor is for the closure studs at [ ]$^{a,c}$. This fatigue usage factor is good for 10,400 cycles of the Plant (Unit) Loading and Plant (Unit) Unloading transient as opposed to the 18,300 cycles specified for the NSSS design transients as given in Section 2.2.1. The large usage factor is mainly due to bolt-up stresses and the large number of cycles associated with the Plant (Unit) Loading and Plant (Unit) Unloading transients. BVPS-1 has traditionally operated with no load follow in accordance with base load operation. Therefore, the qualified number of occurrences should be more than sufficient to cover the anticipated service life, conservatively allowing up to one loading and unloading cycle per day for the remainder of the operating license. However, if there is a change in the way that the unit is operated and the number of loadings and unloadings begin to approach 10,400, the closure studs may require additional evaluation for the effects of fatigue usage.

For BVPS-2, the maximum range of stress intensity for the outlet nozzle safe end increases by [ ]$^{a,c}$ ksi and exceeds the applicable ASME Boiler and Pressure Vessel Code, Section III limit of $3S_m$. This result was acceptable based upon simplified elastic-plastic analysis in accordance with Paragraph 3228.3 of ASME Section III. The maximum range of stress intensity for the bottom head instrumentation tubes was found to exceed $3S_m$ and was justified by simplified elastic-plastic analysis. The maximum ranges of stress intensity for the CRDM housings and bottom head-to-shell juncture also increase, but the values remain less than the applicable $3S_m$ limits. The evaluations show that for all other limiting locations, the existing design stress analyses remain conservative when the revised operating parameters and design transients are incorporated. The maximum cumulative fatigue usage factors at all of the limiting locations (except for the CRDM housings and bottom head instrumentation tubes) increase somewhat from the previous values. However, the increases are mostly minimal, and all of the cumulative fatigue usage factors except for the closure studs and core support guides remain under the 1.0 limit with significant margin. The greatest increase in usage factor is [ ]$^{a,c}$ for the core support guides. This increase resulted from considering the fatigue curve for vessel shell low alloy steel material at the guide-to-vessel shell interface. The maximum cumulative usage factor in the reactor vessel is in the closure studs. The cumulative usage factor for the closure studs actually exceeds the 1.0 limit when all 18,300 occurrences of the Plant (Unit) Loading and Plant (Unit) Unloading transients as specified for the NSSS design transients in Section 2.2.1 are considered in the calculation. However, the CUF is reduced to [ ]$^{a,c}$ when 14,000 occurrences of Plant (Unit) Loading and Plant (Unit) Unloading are applied. BVPS-2 has traditionally operated with no load follow in accordance with base load operation. Therefore, the qualified number of occurrences appears should be more than sufficient to cover the anticipated service life, conservatively allowing up to one loading and unloading cycle per day for the remainder of the operating license. However, if there is a change in the way that the unit is operated and the number of loadings and unloadings begin to approach 14,000, the closure studs may require additional evaluation for the effects of fatigue usage.

The updated maximum ranges of primary plus secondary stress intensity and maximum cumulative fatigue usage factors for the BVPS-1 and BVPS-2 reactor vessels accounting for the EPU are shown in Table 4.1.1-1A and Table 4.1.1-1B, respectively.

A comparison of the revised seismic and LOCA interface loads to the corresponding loads that were previously considered concluded that the main closure, outlet nozzles and core support guides required additional stress analysis to justify application of the seismic and LOCA loads. The seismic and LOCA loads at these reactor internals interfaces were not previously included in the reactor vessel design. Faulted condition stress calculations for the main closure flanges and studs, the outlet nozzles and the core support pads were performed to justify application of the faulted condition loads. The results of these faulted condition analyses are reported in Table 4.1.1-2A and Table 4.1.1-2B for BVPS-1 and BVPS-2, respectively. For BVPS-1, the general primary membrane stress intensities and the maximum primary membrane plus bending stress intensities are reported. For BVPS-2, the general primary membrane stress intensities are not affected by the faulted condition evaluations, therefore, only the maximum primary membrane plus bending stress intensities are reported.

#### 4.1.1.5 Conclusions

Based upon the satisfactory results of the EPU evaluation for the BVPS-1 and BVPS-2 reactor vessels as previously discussed, the reactor vessels are acceptable for plant operation in accordance with the EPU Project. Considering any combination of the design basis and the EPU NSSS transients for the specified numbers of occurrences, the reactor vessel stress and fatigue analyses and evaluations justify operation with a range of vessel outlet temperature ($T_{hot}$) from 603.9° up to 617.0°F and a range of vessel inlet temperature ($T_{cold}$) from 528.5° up to 543.1°F. Such operation of the reactor vessel is shown to be acceptable in accordance with the applicable Edition of Section III of the ASME Boiler and Pressure Vessel Code (1968 Edition with Addenda through Summer 1970 for BVPS-1 and 1971 Edition with Addenda through Summer 1972 for BVPS-2) and the 1974 Edition (for Faulted Conditions only) for the remainder of the plant operating license.

It is also concluded that as long as the total number of Plant (Unit) Loading and Plant (Unit) Unloading cycles remain under the defined cycle limit 10,400 for BVPS-1 and 14,000 for BVPS-2, the closure studs will remain within acceptable fatigue limits as given in the applicable Edition of Section III of the ASME Boiler and Pressure Vessel Code. For BVPS-1 and BVPS-2, the remaining number of cycles are acceptable since the plants are operated base load and the remaining number of cycles conservatively allows for up to one loading and unloading cycle per day for the remainder of the operating licenses. However, if there is a change in the way that the units are operated and the number of loadings and unloadings begin to approach the defined cycle limit, the closure studs may require additional evaluation for the effects of fatigue usage.

The results and conclusions of the analyses and evaluations performed for reactor vessel structural integrity for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

#### 4.1.1.6 References


1. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Vessels," 1968 Edition with Addenda through the Summer 1970, American Society of Mechanical Engineers, New York, New York.

2. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1974 Edition (for Faulted Conditions only), American Society of Mechanical Engineers, New York, New York.

3. ASME Boiler and Pressure Vessel Code, Section III, "Nuclear Power Plant Components," 1971 Edition with Addenda through Summer 1972, American Society of Mechanical Engineers, New York, New York.

**Table 4.1.1-1A**
**BVPS-1 Maximum Ranges of Primary Plus Secondary Stress Intensity and Maximum Cumulative Fatigue Usage Factors for the Reactor Vessel**

| Location | $P_L + P_b + Q$ Range | $U_C$ |
|---|---|---|
| Outlet Nozzles | Nozzle: [ ]$^{a,c}$ ksi < 3 $S_m$ = 80.1 ksi<br>Support Pad: ---(2) | [ ]$^{a,c}$ < 1.0<br>[ ]$^{a,c}$ < 1.0 |
| Inlet Nozzles | Nozzle: [ ]$^{a,c}$ ksi < 3 $S_m$ = 80.1 ksi<br>Support Pad: ---(2) | [ ]$^{a,c}$ < 1.0<br>[ ]$^{a,c}$ < 1.0 |
| Main Closure Flange Region | | |
| 1. Closure Head Flange | [ ]$^{a,c}$ ksi < 3 $S_m$ = 80.1 ksi | -----(2) |
| 2. Vessel Flange | [ ]$^{a,c}$ ksi < 3 $S_m$ = 80.1 ksi | -----(2) |
| 3. Closure Studs | [ ]$^{a,c}$ ksi < 3 $S_m$ = 118.8 ksi | [ ]$^{a,c}$ (1) < 1.0 |
| CRDM Housings | [ ]$^{a,c}$ ksi < 3 $S_m$ = 69.9 ksi | [ ]$^{a,c}$ < 1.0 |
| Vessel Wall Transition | [ ]$^{a,c}$ ksi < 3 $S_m$ = 80.1 ksi | [ ]$^{a,c}$ < 1.0 |
| Bottom Head Juncture | [ ]$^{a,c}$ ksi < 3 $S_m$ = 80.1 ksi | [ ]$^{a,c}$ < 1.0 |
| Bottom Head Instrumentation Tubes | [ ]$^{a,c}$ ksi < 3 $S_m$ = 69.9 ksi | [ ]$^{a,c}$ < 1.0 |
| Core Support Guides | [ ]$^{a,c}$ ksi < 3 $S_m$ = 69.9 ksi | [ ]$^{a,c}$ < 1.0 |

Notes:

(1) This value considers 10,400 occurrences of the Plant (Unit) Loading and Unloading transients instead of the 18,300 occurrences specified for the NSSS design transients. See text in Section 4.1.1.4 for explanation.

(2) This location is not limiting.

**Table 4.1.1-1B**
**BVPS-2 Maximum Ranges of Primary Plus Secondary Stress Intensity and Maximum Cumulative Fatigue Usage Factors for the Reactor Vessel**

| Location | $P_L + P_b + Q$ Range | $U_C$ |
|---|---|---|
| Outlet Nozzles | Nozzle: [ ]a,c ksi < 3 $S_m$ = 80.1 ksi<br>Safe End: [ ]a,c ksi(2) > 3 $S_m$ = 52.3 ksi<br>Support Pad: ---(3) | [ ]a,c < 1.0<br>---(3)<br>[ ]a,c < 1.0 |
| Inlet Nozzles | Nozzle: [ ]a,c ksi < 3 $S_m$ = 80.1 ksi<br>Safe End: [ ]a,c ksi < 3 $S_m$ = 52.3 ksi<br>Support Pad: ---(3) | [ ]a,c < 1.0<br>---(3)<br>[ ]a,c < 1.0 |
| Main Closure Flange Region<br>1. Closure Head Flange<br>2. Vessel Flange<br>3. Closure Studs | <br>[ ]a,c ksi < 3 $S_m$ = 80.1 ksi<br>---(3)<br>[ ]a,c ksi < 3 $S_m$ = 110.3 ksi | <br>---(3)<br>---(3)<br>[ ]a,c (1) < 1.0 |
| CRDM Housings | [ ]a,c ksi < 3 $S_m$ = 69.9 ksi | [ ]a,c < 1.0 |
| Vessel Wall Transition | [ ]a,c ksi < 3 $S_m$ = 80.1 ksi | [ ]a,c < 1.0 |
| Bottom Head Juncture | Shell: [ ]a,c ksi < 3 $S_m$ = 80.1 ksi | [ ]a,c < 1.0 |
| Bottom Head Instrumentation Tubes | [ ]a,c ksi (2) > 3 $S_m$ = 69.9 ksi | [ ]a,c < 1.0 |
| Core Support Guides | Vessel Wall: [ ]a,c ksi < 3 $S_m$ = 80.1 ksi | [ ]a,c < 1.0 |

Notes:

(1) This value considers 14,000 occurrences of the Plant (Unit) Loading and Unloading transients instead of the 18,300 occurrences specified for the NSSS design transients. See text in Section 4.1.1.4 for explanation.

(2) These values that exceed the $3S_m$ are justified by simplified elastic-plastic analysis in accordance with Paragraph NB-3228.3 in ASME III, Subsection NB (Reference 3).

(3) This location is not limiting.

| Table 4.1.1-2A<br>BVPS-1 Faulted Condition Results for the Reactor Vessel | |
|---|---|
| **Maximum General Primary Membrane Stress Intensity, $P_m$** | |
| Closure Head | $P_m$ = [ ]$^{a,c}$ ksi < $0.7S_U$ = 56.0 ksi |
| Vessel Flange | $P_m$ = [ ]$^{a,c}$ ksi < $0.7S_U$ = 56.0 ksi |
| Closure Studs | $P_m$ = [ ]$^{a,c}$ ksi < $2.4S_m$ = 83.5 ksi |
| **Maximum Primary Membrane plus Bending Stress Intensity, $P_L + P_b$** | |
| Closure Head | $P_L + P_b$ = [ ]$^{a,c}$ ksi < $1.05S_m$ = 84.0 ksi |
| Vessel Flange | $P_L + P_b$ = [ ]$^{a,c}$ ksi <$1.05S_U$ = 84.0 ksi |
| Closure Studs | $P_L + P_b$ = [ ]$^{a,c}$ ksi < $3.6S_m$ = 125.28 ksi |
| Outlet Nozzle Shell<br>Outlet Nozzle | $P_L + P_b$ = [ ]$^{a,c}$ ksi < $1.05S_U$ = 84.0 ksi<br>$P_L + P_b$ = [ ]$^{a,c}$ ksi < $1.05S_U$ = 84.0 ksi |
| Core Support Guides Lugs<br>Core Support Guides Shell | $P_L + P_b$ = [ ]$^{a,c}$ ksi < $3.6S_m$ = 83.88 ksi<br>$P_L + P_b$ = [ ]$^{a,c}$ ksi < $1.05S_U$ = 84.0 ksi |

| Table 4.1.1-2B<br>BVPS-2 Faulted Condition Results for the Reactor Vessel | |
|---|---|
| **Maximum Primary Membrane plus Bending Stress Intensity, $P_L + P_b$** | |
| Closure Head Flange | $P_L + P_b = [\quad]^{a,c}$ ksi < $1.05S_U = 84.0$ ksi |
| Vessel Flange | $P_L + P_b = [\quad]^{a,c}$ ksi < $1.05S_U = 84.0$ ksi |
| Closure Studs | $P_L + P_b = [\quad]^{a,c}$ ksi < $2.4S_m = 83.52$ ksi |
| Outlet Nozzle | $P_L + P_b = [\quad]^{a,c}$ ksi < $1.05S_U = 84.0$ ksi |
| Core Support Pad | $P_L + P_b = [\quad]^{a,c}$ ksi < $3.6S_m = 83.88$ ksi |
| Vessel Shell | $P_L + P_b = [\quad]^{a,c}$ ksi < $1.05S_U = 84.0$ ksi |

## 4.2 REACTOR PRESSURE VESSEL SYSTEM

Analyses and evaluations were performed to assess the impact on the reactor internal components of an EPU for the Beaver Valley Power Station to an NSSS power of 2910 MWt (core power of 2900 MWt).

### 4.2.1 Introduction

The reactor pressure vessel (RPV) system consists of the reactor vessel, reactor internals, and fuel and control rod drive mechanisms. The reactor internal's function is to support and orient the reactor core fuel assemblies and control rod assemblies, absorb control rod assembly dynamic loads, and transmit these and other loads to the reactor vessel. The reactor vessel internal components also function to direct coolant flow through the fuel assemblies (core), to provide adequate cooling flow to the various internals structures, and to support in-core instrumentation. They are designed to withstand forces due to structure deadweight, pre-load of fuel assemblies, control rod assembly dynamic loads, vibratory loads, and earthquake accelerations.

Operating a plant at conditions (power and temperature) other than those considered in the original design requires that the reactor vessel system/fuel interface be thoroughly addressed in order to confirm compatibility, and that the structural integrity of the reactor vessel/internals/fuel system is not adversely affected. In addition, thermal-hydraulic analyses are required to determine plant specific core bypass flows, pressure drops, and upper head temperatures in order to provide input to the LOCA and non-LOCA safety analyses, as well as NSSS performance evaluations.

Generally, the areas that are potentially most affected by changes in system operating conditions are:

- Reactor internals system thermal/hydraulic performance
- Rod control cluster assembly (RCCA) scram performance
- Reactor internals system structural response and integrity

The major components and features of the reactor internals system are summarized as follows. The lower core support assembly consists of the lower support plate, lower support columns, and lower core plate and core barrel, and supports the fuel assemblies on the sides and at the bottom. The guidance and alignment of the lower core support assembly during insertion into the reactor vessel is provided by the radial support system and the head-vessel alignment pins, and special temporary guide studs attached to the vessel. The hold-down spring rests on top of the flange of the lower core support assembly. The upper core support assembly consists of the upper support plate, upper support columns, and upper core plate, and rests on top of the hold-down spring. The guidance and alignment of the upper core support assembly, during its insertion, is provided by the head-vessel alignment pins, the upper core plate alignment pins in the core barrel assembly, and the special temporary guide studs attached to the vessel. The alignment of the core, i.e., each fuel assembly, is provided through the engagement of the lower core plate fuel pins into the bottom of the fuel assemblies and the upper core plate fuel pins into the top of the fuel assemblies. The vessel upper head compresses the hold-down spring, providing joint preload.

The core barrel, which is part of the lower core support assembly, provides a flow boundary for the reactor coolant. When the primary coolant enters the reactor vessel, it impinges on the side of the core barrel and is directed downward through the annulus formed by the gap between the outside diameter of the core barrel and the inside diameter of the vessel. The flow then enters the lower plenum area between the bottom of the lower support plate and the vessel bottom head and is redirected upward through the core. After passing through the core, the coolant enters the upper core support region and then proceeds radially outward through the reactor vessel outlet nozzles. The perforations in the various components, such as the lower support plate, control and meter the flow through the core.

The purpose of this section is to summarize the work performed to assess the effect on the reactor pressure vessel/internals system due to an EPU to a core power of 2900 MWt.

### 4.2.2 Input Parameters and Assumptions

The principal input parameters utilized in the analysis of the reactor internal components and RPV system are the NSSS design parameters developed for the EPU provided in Section 2.1.1. For structural analysis evaluations, the NSSS design transients provided in Section 2.2.1 were considered. The fuel considered is a full core of Robust Fuel Assembly (RFA) fuel (including RFA-2) with Intermediate Flow Mixing (IFM) grids and with thimble plugs removed.

## 4.2.3 Description of Analyses and Evaluations

Westinghouse has performed evaluations to assess the effect of the EPU on the reactor pressure vessel/internals system of the Beaver Valley Power Station.

### 4.2.3.1 Thermal-Hydraulic System Evaluations

**System Pressure Losses**

The principal RCS flow route through the reactor pressure vessel system begins at the inlet nozzles. At this point, flow turns downward through the reactor vessel/core barrel annulus. After passing through this downcomer region, the flow enters the lower reactor vessel dome region. This region is occupied by the internals energy absorber structure, lower support columns, bottom-mounted instrumentation columns, and supporting tie plates. From this region, flow passes upward through the lower core plate, and into the core region. After passing up through the core, the coolant flows into the upper plenum, turns, and exits the reactor vessel through the three outlet nozzles. Note that the upper plenum region is occupied by support columns and RCCA guide columns.

A key area in evaluation of core performance is the determination of hydraulic behavior of coolant flow within the reactor internals system, i.e., vessel pressure drops, core bypass flows, RPV fluid temperatures, hydraulic lift forces, and baffle joint momentum flux. The pressure loss data is necessary input to the LOCA and non-LOCA safety analyses and to overall NSSS performance calculations. The hydraulic forces are considered in the assessment of the structural integrity of the reactor internals, core clamping loads generated by the internals hold-down spring, and the stresses in the reactor vessel closure studs.

The THRIVE computer code was used to perform this evaluation by solving the mass and energy balances for the reactor internals fluid system. This THRIVE analysis determined the distribution of pressure and flow within the reactor vessel, internals, and the reactor core. Results were obtained with a full core of RFA fuel (including RFA-2) with IFM grids, thimble plugs removed, and at RCS conditions as summarized in Table 4.2-1. The reactor vessel/internals/fuel pressure drops that occur for these cases are presented in Table 4.2-2.

**Bypass Flow**

<u>Description of Analysis</u>

Bypass flow is the total amount of reactor coolant flow bypassing the core region and is not considered effective in the core heat transfer process. Variations in the size of some of the bypass flow paths, such as gaps at the outlet nozzles and the core cavity, occur during manufacturing or change due to fuel assembly changes. Plant-specific, as-built dimensions are used in order to demonstrate that the bypass flow limits are not violated. Therefore, analysis is performed to estimate core bypass flow values to either show that the design bypass flow limit for the plant will not be exceeded or to determine a revised design core bypass flow.

The present design maximum core bypass flow limit is 6.5% of the total reactor vessel flow with the elimination of thimble plugging devices. The purpose of this evaluation is to show that the design maximum value of 6.5% can be maintained at the RCS conditions of Table 4.2-1. The principal core bypass flow paths are described in the following paragraphs.

Baffle-Barrel Region

The current reactor vessel internals configuration incorporates upward coolant flow in the region between the core barrel and the baffle plates. In this configuration, a majority of the coolant exits the reactor vessel inlet nozzle and flows downward in the annulus between the vessel and core barrel. The downward flow passes over the thermal shield for BVPS-1 or neutron pads for BVPS-2 to the lower plenum, turns, and flows up through the core region. A portion of this flow enters the baffle-barrel region, which consists of vertical baffle plates that follow the periphery of the core. These are joined to the core barrel by horizontal former plates spaced along the elevation of the baffle plates. All of the former plates have flow holes machined in them. Some flow from the lower plenum enters the baffle-barrel region at the lower core plate and exits above the core. There will also be some flow exchange between the baffle-barrel region and the core through the baffle plate gaps.

Vessel Head Cooling Spray Nozzles

These nozzles are flow paths between the reactor vessel and core barrel annulus and the fluid volume in the vessel closure head region above the upper support plate. A fraction of the flow that enters the vessel inlet nozzles and into the vessel/barrel downcomer passes through these nozzles and into the vessel closure head region. The purpose of these flow paths is to allow circulation of a small fraction of the cold leg coolant into the upper head region of the reactor vessel.

Core Barrel – Reactor Vessel Outlet Nozzle Gap

Some of the flow that enters the vessel/barrel downcomer leaks through the gaps between the core barrel outlet nozzles and the reactor vessel outlet nozzles and merges with the vessel outlet nozzle flow. Since the lower reactor internals are designed to be removable from the reactor vessel, a small circumferential gap exists at each of the outlet nozzle locations. While the gap is designed to be very small and closes down somewhat at operating conditions due to the differential coefficient of thermal expansion between the reactor internals and the reactor vessel, there is some amount of flow which leaks directly from the vessel inlet/downcomer region and out through these nozzle gaps.

Fuel Assembly – Baffle Plate Cavity Gap

The baffle plates surround the reactor fuel assemblies or core region. The gap between the peripheral fuel assemblies and the baffle plates is referred to as the core cavity region. This is the core bypass flow path between the peripheral fuel assemblies and the core baffle plates.

Fuel Assembly Thimble Tubes

Thimble tubes are used as paths for the insertion and removal of control rods, thimble plugging devices, and various core components such as burnable absorbers. These tubes are physically part of each fuel assembly and flow within them is partially effective in removing core heat. However, such flow is analytically not considered to be effective in heat removal, and is consequentially considered to be part of the core bypass flow.

Bypass Flow Analysis Results

Fuel assembly hydraulic characteristics and system parameters, such as inlet temperature, reactor coolant pressure, and flow were used in conjunction with the THRIVE code to determine the impact of EPU RCS conditions on the total core bypass flow. The calculated core bypass flow value is 5.74% for BVPS-1 and 5.63% for BVPS-2 at the RCS conditions of Table 4.2-1. Therefore, the design maximum core bypass flow value of 6.5% of the total vessel flow can be maintained.

**Upper Head Fluid Temperatures**

The average temperature of the primary coolant fluid that occupies the reactor vessel closure head volume is an important initial condition for certain dynamic LOCA analyses. Therefore, it was necessary to determine the upper head temperature when changes in the RCS conditions take place in the plant. Determination of upper head temperature stemmed from the THRIVE analysis used to assess the core bypass flow. The THRIVE code models the interaction between all different flow paths into and out of the closure head region. Based on this interaction, it calculates the core bypass flow into the head region and average head fluid temperature for different flow path conditions. The upper head mean fluid temperatures are provided in Table 4.2-3 for the cases shown in Table 4.2-1.

**Hydraulic Lift Forces**

An evaluation was performed to estimate hydraulic lift forces on the various reactor internal components for the EPU parameters shown in Table 4.2-1. This was done to show that the reactor internals assembly would remain seated and stable for all conditions. If the impact of the EPU changes on lift forces is found to be significant, then the estimated hydraulic lift forces would be combined with other mechanical and body forces to evaluate the resultant pre-load of the core barrel flange against the reactor vessel. Table 4.2-4 presents comparisons of hydraulic lift forces on various reactor internal components for the cases shown in Table 4.2-1. The total hydraulic lift forces on the lower internals package were slightly increased compared to present analyzed conditions. Based on the evaluation, the reactor internals will remain seated and stable for the EPU RCS conditions.

**Baffle Joint Momentum Flux and Fuel Rod Stability**

Baffle jetting is a hydraulically induced instability or vibration of fuel rods caused by a high velocity jet of water. This jet is created by high pressure water being forced through gaps between the baffle plates which surround the core. The baffle jetting phenomenon could lead to fuel cladding damage.

With the "converted upflow" baffle barrel region configuration of BVPS-1 or the standard upflow baffle barrel region configuration of BVPS-2, the momentum flux margins remain acceptable.

#### 4.2.3.2 Flow-Induced Vibrations

Flow-induced vibrations of pressurized water reactor internals have been studied for a number of years. The objective of these studies is to show the structural integrity and reliability of reactor internal components. These efforts have included in-plant tests, scale-model tests, as well as tests in fabricators' shops and bench tests of components, along with various analytical investigations. The results of these scale-model and in-plant tests indicate that the vibrational behavior of two-, three-, and four-loop plants is essentially similar, and the results obtained from each of the tests compliment one another and make possible a better understanding of the flow-induced vibration phenomena.

Based on the analysis, the Beaver Valley Power Station reactor internals response due to flow-induced vibrations is extremely small and well within the allowable based on the high cycle endurance limit for the material.

#### 4.2.3.3 Structural Evaluation of Reactor Internal Components

In addition to supporting the core, a secondary function of the reactor vessel internals assembly is to direct coolant flows within the vessel. While directing the primary flow through the core, the internals assembly also establishes secondary coolant flow paths for cooling the upper regions of the reactor vessel and for cooling the internals structural components. Some of the parameters influencing the mechanical design of the internals lower assembly are the pressure and temperature differentials across its component parts and the flow rate required to remove the heat generated within the structural components due to radiation (e.g., gamma heating). The configuration of the internals provides for adequate cooling capability.

Structural evaluations are performed to demonstrate that the structural integrity of the reactor components is not adversely affected directly by the change in RCS conditions and transients and/or by secondary effects of the change on reactor thermal-hydraulic or structural performance. The presence of heat generated in reactor internal components, along with the various fluid temperatures, results in thermal gradients within and between components. These thermal gradients result in thermal stresses and thermal growth, which must be accounted for in the design and analysis of the various components.

**Design Transients**

Section 2.2.1 provides a listing of the normal and upset NSSS design transients for the primary loop. In order to assess the effect of these transients, the transients were grouped and enveloped (e.g., Upset Up and Upset Down). "Up" indicates an increasing temperature during the transient, while "Down" indicates a decreasing temperature. The process of enveloping transient groups consists of taking the largest temperature change (in one direction only) in the shortest time span of all the transients in the group. If more margin is needed for a component stress analysis or if a particular transient is too severe to be included in an envelope, then it is analyzed separately.

After the enveloping of the revised design transients was completed, the new enveloping curves were then used in the structural evaluations of the reactor internal components.

**Component Evaluations**

This section summarizes the results of structural evaluations performed for the key reactor internals components at the EPU RCS conditions. Westinghouse performed a review and an evaluation of the effects of the NSSS design transients and the EPU on the following reactor internal components:

- Lower Core Plate
- Lower Core Support Plate
- Lower Support Columns
- Core Barrel Assembly
- Lower Radial Supports
- Upper Core Plate
- Upper Core Plate Alignment Pins
- Upper Support Assembly
- Baffle/Former Bolts

The stresses, margins of safety, and cumulative fatigue usage factors for some of the reactor internals components are provided in Table 4.2-5 which shows the effects of the EPU.

**Baffle-Barrel Region Evaluations**

The baffle-barrel region consists of a core barrel into which baffle plates are installed, supported by bolting interconnecting former plates to the baffle and core barrel. The baffle-to-former bolts restrain the motion of the baffle plates that surround the core. These bolts are subjected to primary stresses consisting of deadweight, hydraulic pressure differentials, seismic and LOCA loads, as well as secondary stresses consisting of bolt preload and thermal loads resulting from RCS temperatures and gamma heating rates. The EPU does not affect the deadweight or seismic loads. The baffle-to-former bolt thermal loads are induced by differences in the average metal temperature between the core barrel and baffle plate. The difference in temperature between the core barrel and baffle has been determined for the EPU.

For the EPU, the gamma heating rates seen by the baffle-barrel region would increase proportionally with the increase in power. The effect of this increase in gamma heating rates and the effect of the core power distributions and design transients produces a difference in temperature between the baffle and core barrel. The effect of core power distributions offset the increased loads due to the gamma heating rates and design transients, resulting in a smaller temperature difference. Therefore, the baffle-to-barrel temperature difference for the EPU was less than that from previous evaluations, so the baffle-former bolt displacements remain less than those previously evaluated.

**Lower Core Plate Structural Analysis**

The lower core plate is a perforated circular plate that supports and positions the fuel assemblies. The plate contains numerous holes to allow fluid flow through the plate. The fluid flow is provided to each fuel assembly and the baffle-barrel region. The plate is bolted at the periphery to a ring welded to the inside diameter of the core barrel. The center span of the plate is supported by the lower support columns, which are attached at the lower end to the lower support plate.

A structural evaluation was performed to demonstrate that the EPU does not adversely affect the structural integrity of the lower core plate. The EPU causes an increase in the heat generated within the lower core plate. The revised design transients for the EPU were evaluated. Heat generation rates were developed specifically for this evaluation.

The conclusion of the evaluation is that the structural integrity of the lower core plate is maintained. The new RCS conditions, which are due to the EPU, produced acceptable stresses, margins of safety, and fatigue utilization factor. See Table 4.2-5.

**Upper Core Plate Structural Analysis**

The upper core plate positions the upper ends of the fuel assemblies and the lower ends of the control rod guide tubes, thus serving as the transitioning member for the control rods in entry and retraction from the fuel assemblies. It also controls coolant flow in its exit from the fuel assemblies and serves as a boundary between the core and the exit plenum. The upper core plate is restrained from vertical movement by the upper support columns, which are attached to the upper support plate assembly. Four equally spaced core plate alignment pins restrain the lateral movement.

A structural evaluation was performed to demonstrate that the EPU does not adversely affect the structural integrity of the upper core plate. The EPU causes an increase in the heat generation seen by the upper core plate. The design transients developed for the EPU were evaluated. Heat generation rates were developed specifically for this evaluation.

The conclusion of the evaluation is that the structural integrity of the upper core plate is maintained. The new RCS conditions, which are due to the EPU, produced acceptable stresses, margins of safety, and fatigue utilization factor. See Table 4.2-5.

#### 4.2.3.4 RCCA Drop Time Evaluation

The RCCAs represent perhaps the most critical interface between the fuel assemblies and the other internal components. It is imperative to show that the EPU RCS conditions will not adversely impact the operation of the RCCAs, either during accident conditions or normal operation.

In general, a plant-specific RCCA drop time performance assessment involves the following steps:

1. Obtain actual plant drop time-to-dashpot entry data at no-flow and full-flow conditions for each RCCA location.

2. Develop an analytical model of the plant's driveline configuration and system operating conditions corresponding to those measurements. A driveline is considered to be that subset of components affecting RCCA drop time. These components are the fuel, upper core plate, upper and lower guide tubes, upper support plate, reactor closure head penetration, thermal sleeve, CRDM, rod travel housing, and the RCCA/drive rod assembly. The system operating conditions include temperature, pressure, and flow. The analytical model consists of values for parameters that describe geometry of driveline components, component mechanical interaction relationships,

hydraulic resistances of flow paths, RCCA/drive rod assembly weight, and system operating conditions.

3. Use a coded algorithm previously developed by Westinghouse, with the analytical model, to correlate the model to the plant measured drop times. This algorithm has been used for this analysis since the original plant design. The algorithm solves Newton's second law of motion. This law states:

$$\Sigma F = (W/g) \times (dV/dt)$$

where:

| | | |
|---|---|---|
| $\Sigma F$ | = | Sum of various forces acting on the RCCA/drive rod assembly at any time, (t) |
| W | = | total weight of RCCA/drive rod assembly |
| g | = | acceleration due to gravity, (32.2 ft/sec$^2$) |
| V | = | assembly velocity, (ft/sec) |
| t | = | drop time after CRDM latch release of drive rod, (sec) |

The correlation involves adjustment of specific code input parameters:

a. Characterize RCCA drop performance from no-flow (0%) through full-flow (100%) based on no-flow and full-flow core average drop time measurements, and

b. Isolate and account for the effects of variations in drive line mechanical interference drag force under normal conditions, and variations in drive line flows across the core, based on core-maximum drop time measurements at no-flow and full-flow, respectively.

Adjust the model to account for the new system operating conditions being considered due to the EPU. Also, conservatively account for:

a. Component geometric design tolerances

b. Hydraulic performance uncertainties (related to fuel assembly hydraulic resistance, guide tube/RCCA wear, and reactor coolant flow rate)

c. Abnormal environmental conditions (particularly seismic events)

4. Assess the impact of such changes in driveline components and/or primary system operating conditions on the limiting RCCA drop time characteristics used in the plant accident analyses. These limiting characteristics are the most severe drop time-to-dashpot entry and normalized RCCA drop time position-versus-time relationship estimated based on the tolerances, uncertainties, and abnormal environmental conditions identified above.

The analysis performed determined the potential impact of the conditions shown in Table 4.2-1 on the limiting RCCA drop time. The maximum estimated RCCA drop time with the seismic allowance was calculated to be 2.3 seconds to the top of dashpot, which is still less than the current Technical Specification limit of 2.7 seconds.

#### 4.2.3.5 LOCA and Seismic Evaluations

**LOCA Analyses**

To perform the RPV LOCA analyses for the Beaver Valley Power Station, a finite element model of the RPV system is developed.

The mathematical model of the RPV is a three-dimensional nonlinear finite element model that represents the dynamic characteristics of the reactor vessel and its internals in the six geometric degrees of freedom. The model was developed using the WECAN computer code. The WECAN computer code (or predecessor codes) has been used for this analysis since the original plant design.

The WECAN computer code, which is used to determine the response of the reactor vessel and its internals, is a general purpose finite element code. In the finite element approach, the structure is divided into a finite number of members or elements. The inertia and stiffness matrices, as well as the force array, are first calculated for each element in the local coordinates. Employing appropriate transformation, the element global matrices and arrays are then computed. Finally, the global element matrices and arrays are assembled into the global structural matrices and arrays, and used for dynamic solution of the differential equation of motion for the structure.

In order to evaluate the impact of changes in RCS conditions on the dynamic response of the RPV system, LOCA analyses were performed to generate core plate motions and the reactor vessel/internals interface loads. The core plate motions are then used to evaluate the structural integrity of the core. Since application of leak-before-break (LBB) methodology has been licensed for the main coolant loop, consideration of breaks in the main coolant loop are not required for structural evaluations. The next limiting breaks to be considered are the branch line breaks. The hydraulic LOCA forces that are used in the reactor vessel LOCA analysis are for breaks in the 12" accumulator line (cold leg) and the 14" residual heat removal line (hot leg) for BVPS-1 and for breaks in the 4" line (cold leg) and the 3" line (hot leg) for BVPS-2.

Following a postulated LOCA, forces are imposed on the reactor vessel and its internals. These forces result from the release of the pressurized primary system coolant and, for auxiliary pipe breaks, from the disturbance of the mechanical equilibrium in the piping system prior to the rupture. The release of pressurized coolant results in traveling depressurization waves in the primary system. These depressurization waves are characterized by a wavefront with low pressure on one side and high pressure on the other. The wavefront translates and reflects throughout the primary system until the system is completely depressurized. The rapid depressurization results in transient hydraulic loads on the mechanical equipment of the system.

The LOCA loads applied to the reactor pressure vessel system consist of: (1) reactor internal hydraulic loads (vertical and horizontal), and (2) reactor coolant loop mechanical loads. All the loads are calculated individually and combined in a time history manner.

The MULTIFLEX computer code (Reference 1) calculates the hydraulic transients within the entire primary coolant system. It considers sub-cooled, transition, and two-phase (saturated) blowdown regimes. The MULTIFLEX program employs the method of characteristics to solve the conservation laws, and assumes one-dimensionality of flow and homogeneity of the liquid-vapor mixture.

The MULTIFLEX code considers a coupled fluid-structure interaction by accounting for the deflection of constraining boundaries, which are represented by separate spring-mass oscillator systems. A beam model of the core support barrel has been developed from the structural properties of the core barrel. In this model, the cylindrical barrel is vertically divided into various segments and the pressure/wall motions are projected onto the plane parallel to the inlet nozzle on the loop with the postulated auxiliary line pipe break. Horizontally, the barrel is divided into 10 segments; each segment consists of 3 separate walls. The spatial pressure variation at each time step is transformed into 10 horizontal forces, which act on the 10 mass points of the beam model. Each flexible wall is bounded on either side by a hydraulic flow path. The motion of the flexible walls is determined by solving the global equations of motion for the masses representing the forced vibration of an undamped beam.

The severity of a postulated break in a reactor vessel is related to two factors: the distance from the reactor vessel to the break location and the break opening area. The nature of the reactor vessel decompression following a LOCA, as controlled by the internals structural configuration previously discussed, results in larger reactor internal hydraulic forces for pipe breaks in the cold leg than in the hot leg (for breaks of similar area and distance from the RPV). Pipe breaks farther away are less severe because the pressure wave attenuates as it propagates toward the reactor vessel. Therefore, pipe breaks at the reactor vessel inlet nozzle are more severe, because of the absence of pressure wave attenuation and the structural configuration of the core. In general, the auxiliary line breaks like the 12" accumulator line and the 14" residual heat removal line breaks for BVPS-1 and the 4" line and 3" line breaks for BVPS-2 are not as severe as the main line breaks such as RPV inlet nozzle or reactor coolant pump outlet nozzle break.

The results of reactor vessel displacements and the impact forces calculated at vessel/internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals.

The core plate motions for both breaks were provided for use in fuel grid crush analysis and to confirm the structural integrity of the fuel.

**Seismic Analyses**

The non-linear time history seismic analyses of the reactor pressure vessel system includes the development of the system finite element model and the synthesized time history accelerations.

Similar to the response during LOCA, the reactor pressure vessel system seismic model included sub-models of the reactor vessel, nozzles, internals, fuel and control rod drive mechanism. The WECAN finite element model described for LOCA was modified to include the fluid-structure interaction in the reactor pressure vessel model for the seismic OBE and SSE time history evaluations. The WECAN reactor vessel/internals/fuel assembly model incorporated the effects of fluid-structure interaction in the down-comer region via hydrodynamic mass matrices between two concentric cylinders (between the core barrel and reactor vessel). The fluid-structure interaction in the seismic analysis is different from that included in the LOCA analysis. In the LOCA analysis, the fluid-structure interaction is included through the MULTIFLEX code; whereas in the seismic analysis, the fluid-structure interaction in the downcomer region (between the core barrel and reactor vessel) is incorporated through the hydrodynamic mass matrices. The mass matrices with off-diagonal terms are incorporated between nodes on the core barrel and reactor vessel shell.

For a time history response of the reactor pressure vessel and its internals under seismic excitation, synthesized time history accelerations are required. The synthesized time history accelerations used in the reactor pressure vessel system analysis were based on the applicable Beaver Valley Power Station response spectra.

The results of the system seismic analysis include time history displacements and impact forces for all the major components. The reactor vessel displacements and the impact forces calculated at vessel/internals interfaces are used to evaluate the structural integrity of the reactor vessel and its internals. The core plate motions were provided for use in fuel grid crush analysis and to confirm the structural integrity of the fuel.

### 4.2.4 Acceptance Criteria and Results

The main applicable criteria and results are presented for the areas evaluated:

- Thermal-Hydraulic Performance:

  The THRIVE analysis is performed to evaluate the thermal-hydraulic performance of the reactor pressure vessel system. The THRIVE code computes the vessel pressure drops, core bypass flows, RPV fluid temperatures, hydraulic lift forces, and baffle joint momentum flux. The THRIVE outputs serve as inputs to other analyses including the reactor internals structural analysis, LOCA, and non-LOCA analyses.

  The acceptance criteria for design maximum core bypass flow is 6.5%. The calculated core bypass flow value is 5.74% for BVPS-1 and 5.63% for BVPS-2 at the RCS conditions of Table 4.2-1.

- Flow-Induced Vibration Response:

  The flow-induced vibration (FIV) response of reactor internal components, in general, depends upon reactor vessel inlet flow rates (such as mechanical design flow), reactor vessel inlet temperature and reactor vessel outlet temperature. The response of the lower internals (core barrel) depends on the vessel inlet temperature and the inlet flow rates. The response of the upper internals (guide tubes and upper support columns) depends on the vessel outlet temperature and the flow exiting through the outlet nozzles. The acceptance criteria for the flow-induced vibration response are that the stresses from the FIV amplitudes remain within the endurance limit of the material for high cycle fatigue and that component loads are within acceptable limits. The reactor internals response due to flow-induced vibrations for EPU conditions is extremely small and well within the allowable based on the high cycle endurance limit for the material.

- Structural Adequacy of Reactor Internal Components:

  The design of the Beaver Valley Power Station reactor internals was evaluated according to Westinghouse internal criteria that are similar to Subsection NG of the ASME B&PV, Section III. The Beaver Valley Power Station internals were designed prior to the introduction of Subsection NG of the ASME B&PV Code Section III. The structural evaluations performed demonstrate that the structural integrity of the reactor components is not adversely affected at the EPU conditions.

- Control Rod Drop Analysis:

  The rod drop time values generated, consistent with plant operating parameters and configuration, should be within the limit defined in the Technical Specifications which is 2.7 seconds. The current calculated RCCA drop time is 2.3 seconds.

- LOCA and Seismic Analyses of Reactor Vessel and Internals:

  The beam data serves as input to the generation of LOCA hydraulic forces. The interface loads and the time history nodal displacements of the reactor internals components, determined in the LOCA and seismic analyses, serve as inputs to various structural analyses of the internals, fuel, and the vessel.

### 4.2.5 Conclusions

All structural and thermal-hydraulic analyses and evaluations conclude that the reactor pressure vessel system is acceptable for operation at the EPU conditions.

The results and conclusions of the analyses and evaluations performed for the reactor pressure vessel system for the reactor power of 2900 MWt (2910 MWt NSSS power) bound and support operation at the current reactor power of 2689 MWt (2697 MWt NSSS power), thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

### 4.2.6 References

1. WCAP-8708-P/A, MULTIFLEX, a FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic Structure System Dynamics, 9/77, (Westinghouse Proprietary).

| Table 4.2-1<br>Reactor Coolant System (RCS) Conditions for EPU[1] | | | |
|---|---|---|---|
| **Case** | **Core Power Level (MWt)** | **Inlet Temp. (°F)** | **Thermal Design Flow/Pressure** |
| 1&2 | 2900 | 528.5 | 87,200 gpm/loop<br>2250 psia |
| 3&4 | 2900 | 543.1 | 87,200 gpm/loop<br>2250 psia |

Note:

(1) These parameters are taken from Section 2.1.1.

| Table 4.2-2<br>Reactor Internals Coolant Pressure Drop Comparison for EPU (psi)[1] | | | | | | |
|---|---|---|---|---|---|---|
| **Component** | **BVPS-1** | | | **BVPS-2** | | |
| | **Nominal[2]** | **Case 1&2** | **Case 3&4** | **Nominal[2]** | **Case 1&2** | **Case 3&4** |
| Inlet Nozzle | 7.01 | 6.92 | 6.80 | 5.89 | 5.82 | 5.71 |
| Barrel/Vessel Annulus | 1.82 | 1.80 | 1.76 | 0.44 | 0.43 | 0.42 |
| Lower Plenum | 4.66 | 4.61 | 4.52 | 4.63 | 4.58 | 4.50 |
| Lower Support Plate | 0.75 | 0.74 | 0.73 | 0.72 | 0.71 | 0.70 |
| Diffuser Plate | 0.90 | 0.89 | 0.87 | N/A[4] | N/A[4] | N/A[4] |
| Core[3] | 18.41 | 18.47 | 18.21 | 18.41 | 18.47 | 18.21 |
| Upper Plenum | 0.40 | 0.40 | 0.39 | 0.40 | 0.40 | 0.39 |
| Outlet Nozzle | 1.71 | 1.69 | 1.68 | 1.28 | 1.27 | 1.26 |
| TOTAL PRESSURE DROP | 35.65 | 35.52 | 34.97 | 31.77 | 31.68 | 31.19 |

Notes:

(1) Pressure drops calculated based on a thermal design flow rate.

(2) Nominal case shows current analysis value for current power level.

(3) Core pressure drop includes lower core plate, fuel and upper core plate pressure drops.

(4) The BVPS-2 reactor internals do not include a diffuser plate.

| Table 4.2-3<br>Mean Upper Head Fluid Temperatures for EPU | | |
|---|---|---|
| **Case No.** | **BVPS-1 Temperatures (°F)** | **BVPS-2 Temperatures (°F)** |
| 1&2 | 587.5 | 587.6 |
| 3&4 | 601.2 | 601.3 |

| Table 4.2-4<br>Comparison of Hydraulic Lift Forces for EPU (lbs)(1) | | | | | | |
|---|---|---|---|---|---|---|
| | **BVPS-1** | | | **BVPS-2** | | |
| **Component** | **Nominal(2)** | **Case 1&2** | **Case 3&4** | **Nominal(2)** | **Case 1&2** | **Case 3&4** |
| Core Barrel | 24796 | 25229 | 24838 | 22539 | 23512 | 23151 |
| Core Barrel Flange | 109369 | 111054 | 109263 | 97577 | 101642 | 100026 |
| Barrel/Vessel Annulus | -7678 | -7724 | -7582 | -833 | -860 | -844 |
| Lower Support Plate | 14049 | 14132 | 13873 | 13172 | 13602 | 13353 |
| Diffuser Plate | 16835 | 16935 | 16625 | N/A(4) | N/A(4) | N/A(4) |
| Lower Core Plate | 9351 | 9017 | 8852 | 9109 | 9017 | 8852 |
| Former Plates | 46122 | 49630 | 48884 | 45032 | 49633 | 48887 |
| Baffle Plates | 12051 | 12968 | 12773 | 11763 | 12965 | 12770 |
| Upper Core Plate(3) | 33728 | 18763 | 18601 | 32907 | 18750 | 18588 |
| Upper Support Plate | 1842 | 1856 | 1841 | 1812 | 1864 | 1848 |
| TOTAL FORCE | 260465 | 251859 | 247967 | 233079 | 230125 | 226631 |

Notes:

(1) Lift forces calculated based on a mechanical design flow rate.

(2) Nominal case shows current analysis value for current power level.

(3) The removal of the fuel thimble plugs causes a reduction in the upper core plate hydraulic lift force.

(4) The BVPS-2 reactor internals do not include a diffuser plate.

| Table 4.2-5<br>Summary of Stresses, Safety Margins, and Cumulative Usage Factors<br>(Effect of EPU) | | | | |
|---|---|---|---|---|
| **Reactor Internal Component** | **$P_m+P_b+Q$** | | | **Cumulative Usage Factor (CUF)** |
| | **Actual Stress (psi)** | **Allowable Stress (psi)** | **Margin of Safety(1)** | |
| Lower Core Plate | [ ]a,c | 48,600 | [ ]a,c | [ ]a,c |
| Upper Core Plate | [ ]a,c | 48,600 | [ ]a,c | [ ]a,c |

Note:

(1) Margin of Safety = (Allowable/Actual) - 1

## 4.4 CONTROL ROD DRIVE MECHANISMS AND CAPPED LATCH HOUSINGS

### 4.4.1 Introduction

This section addresses the ASME Code structural considerations for the pressure boundary components of the full length Model L-106A Control Rod Drive Mechanisms (CRDMs) and Capped Latch Housings (CLHs). The CRDMs and CLHs were evaluated for the EPU Project PCWG parameters and the associated NSSS design transients.

This evaluation provides verification of continued structural suitability of the pressure boundary components of the existing CRDMs and CLHs for the EPU Project.

### 4.4.2 Input Parameters and Assumptions

The BVPS-1 and BVPS-2 Model L-106A CRDMs and CLHs were originally designed and analyzed to the generic component design reports and the ASME Code (see Table 4.4-1).

The input parameters that were used to perform the analyses and evaluations for EPU include the original NSSS design parameters and NSSS design transients, the EPU PCWG parameters (Section 2.1.1) along with the EPU NSSS design transients (Section 2.2.1), and the current design basis evaluations for the CRDMs and CLHs.

### 4.4.3 Description of Analyses and Evaluations

The ASME Code structural and fatigue limits and criteria of the generic CRDM and CLH reports are used to define the basis of the adequacy in the current evaluations of the Beaver Valley Power Station. The ASME Code year and addenda for each of the reports are shown in Table 4.4-1.

The CRDMs and CLHs are installed in the reactor vessel upper head (hot head CRDMs) and are affected by the PCWG Reactor Coolant Pressure, PCWG Vessel Outlet Temperature, and the hot leg NSSS design transients. The PCWG Reactor Coolant Pressure is the same for both the EPU Project and for the current basis, which did not change from the original analysis. Since the PCWG Reactor Coolant Pressure remains the same as originally specified, the CRDMs and CLHs remain bounded for the PCWG Reactor Coolant Pressure condition for EPU.

The highest PCWG Vessel Outlet Temperature for the EPU Project is 617°F. Since most of the previous analyses used material allowables based on the Design Temperature of 650°F, the revised PCWG temperatures defined for the EPU Project are, in most cases, enveloped by the previous analyses. The evaluations that were not enveloped by prior work are the bell mouthing analysis for the upper threaded joint area, which used a material allowable based on a local temperature calculated for one of the transients, and the fatigue analyses for the CRDMs and the CLHs, which were revised for the EPU NSSS design transients.

**CRDM Evaluation**

The BVPS-1 and BVPS-2 CRDMs are designed to the requirements of the Westinghouse equipment specification. The original CRDM analysis was performed to Section III of the ASME Code, the 1971 Edition with Addenda through the Winter of 1971 for BVPS-1 and the 1971 Edition with Addenda through the Winter of 1973 for BVPS-2. Additional calculations were performed to demonstrate that the upper joint configuration of the CRDMs designed prior the Winter of the 1971 ASME Code conformed to the requirements of Winter 1969 Addenda to the 1968 Edition of the code. For BVPS-2, evaluations were also performed for replacement of part length CRDMs and the acceptability of a spare CRDM. The code of record for these evaluations for BVPS-2 is included in Table 4.4-1.

A comparison of the temperatures and design transients for the original analysis to those for the EPU showed that since most of the previous analyses used material allowables based on the Design Temperature of 650°F, the revised PCWG temperatures defined for the EPU Project are, in most cases, enveloped by the previous analyses. The evaluations that were not enveloped by prior work are the bell mouthing analysis for the upper threaded joint area, which used a material allowable based on a local temperature calculated for one of the transients, and the fatigue analyses for the CRDMs, which were revised for the EPU design transients.

**CLH Evaluation**

The BVPS-1 and BVPS-2 CLHs are designed to the requirements of the Westinghouse equipment specification. The original CLH analysis was performed to Section III of the ASME Code, 1974 Edition with Addenda through Winter 1975.

A comparison of the temperatures and design transients for the original analysis to those for the EPU showed that only the fatigue analysis for the CLH cap required further analysis. The revised analysis resulted in a lower fatigue usage factor due to the fact that the previous analysis was based on very conservative parameters and transients.

### 4.4.4 Acceptance Criteria and Results

The acceptance criteria and results of the EPU analyses and evaluations for the CRDMs and CLHs are summarized below:

- CRDMs – Bell Mouthing Analysis for the Upper Joint Threaded Area

  Calculated Stress Intensity = [ ]$^{a,c}$ psi
  Allowable Stress Intensity = 21,010 psi

- CRDMs – Fatigue Analysis for the Upper Joint Canopy

  Usage Factor U = [ ]$^{a,c}$
  Allowable Usage Factor < 1.0

- CLHs – Fatigue Analysis for the CLH Cap

  Usage Factor U = [ ]$^{a,c}$
  Allowable Usage Factor < 1.0

### 4.4.5 Conclusions

The BVPS-1 and BVPS-2 CRDMs and CLHs were evaluated for the EPU PCWG parameters (Section 2.1.1) and the associated NSSS design transients (Section 2.2.1). In most cases, the existing analyses and evaluations remained applicable and bounding. Where this was not the case, new calculations were performed for the limiting components and the results evaluated to establish the structural acceptability of the CRDM and CLH pressure boundary components in accordance with the ASME Code.

Based on the previous analyses and the analyses and evaluations performed for EPU, the BVPS-1 and BVPS-2 CRDM and CLH pressure boundary components are acceptable in accordance with the ASME Code for the EPU Project.

The results and conclusions of the analyses and evaluations performed for CRDMs and CLHs for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

**Table 4.4-1**
**CRDM and CLH Component Evaluation Reports and ASME Code Editions**

| Report | ASME Code Edition |
|---|---|
| Generic CRDM Analysis (BVPS-1 and BVPS-2) | 1971 Edition with Addenda through Winter 1971 |
| Generic CRDM Upper Joint Analysis (BVPS-1 and BVPS-2) | 1968 Edition with Addenda through Winter 1969 |
| Generic CLH Analysis (BVPS-1 and BVPS-2) | 1974 Edition with Addenda through Winter 1975 |
| CRDM/CLH Analysis (BVPS-2) | 1971 Edition with Addenda through Winter 1973 |
| Replacement of Part Length CRDMs (BVPS-2) | 1971 Edition with Addenda through Winter 1973 |
| Spare CRDM (BVPS-2) | 1971 Edition with Addenda through Winter 1973 |
| Generic CRDM Analysis (BVPS-2) | Winter 1974 |
| Generic CRDM/CLH Analysis for COMS Transient (BVPS-2) | 1983 Edition with Addenda through the 1986 Edition |

## 4.5 REACTOR COOLANT LOOP PIPING AND SUPPORTS

The EPU Project included analyses and evaluations for the reactor coolant loop piping and supports, including pressurizer surge line stratification and application of leak-before-break (LBB) methodology. The analyses and evaluations for pressurizer surge line stratification and the application of LBB methodology are addressed in this section. The analyses and evaluations for the reactor coolant loop piping and supports are addressed in Section 8.3.

The EPU considered the potential small increase in PWSCC susceptibility of Alloy 600 components in the RCS Piping System. An Alloy 600 management program has been established to manage and identify mitigative actions to address PWSCC of Alloy 600 material in the RCS utilizing site specific assessment of each Alloy 600 and Alloy 82/182 weld location. This program incorporates EPU conditions such that the impact of EPU on Alloy 600 material PWSCC susceptibility is properly managed at BVPS-1 and BVPS-2.

### 4.5.1 Pressurizer Surge Line Stratification

#### 4.5.1.1 Introduction

An evaluation was performed for the BVPS-1 and BVPS-2 pressurizer surge line stratification analysis to address the impact of the EPU power level including revised NSSS design transients and revised design loads for the EPU Project.

#### 4.5.1.2 Input Parameters and Assumptions

The inputs to the pressurizer surge line stratification analysis included the NSSS design (PCWG) parameters for EPU in Section 2.1.1, the revised NSSS design transients for EPU in Section 2.2.1, and the revised design loads for EPU conditions.

#### 4.5.1.3 Description of Analyses and Evaluations

The pressurizer surge line was reanalyzed to address the impact of the EPU Project on the original pressurizer surge line stratification analysis (Reference 1 for BVPS-1 and References 2, 3, 4, and 5 for BVPS-2). The analysis for EPU was consistent with the original pressurizer surge line stratification analysis that was submitted to the NRC.

#### 4.5.1.4 Acceptance Criteria and Results

The results of the evaluation of the pressurizer surge line stratification are shown in Table 4.5.1-1. The ASME Equation 12 and 13 stress and cumulative fatigue usage factor are below allowable limits as established in the ASME Code (Reference 6).

#### 4.5.1.5 Conclusions

The effects of EPU on the BVPS-1 and BVPS-2 pressurizer surge line stratification analysis are found to be within ASME code allowable limits. The pressurizer surge line piping will maintain its structural integrity and meet all stratification analysis requirements at EPU conditions.

The results and conclusions of the analyses and evaluations performed for pressurizer surge line stratification for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

#### 4.5.1.6 References


1. WCAP-12727, "Evaluation of Thermal Stratification for Beaver Valley Unit 1 Pressurizer Surge Line," November 1990.

2. WCAP-12093, "Evaluation of Thermal Stratification for Beaver Valley Unit 2 Pressurizer Surge Line," December 1988.

3. WCAP-12093, Supplement 1, "Additional Information in Support of the Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line," February 1989.

4. WCAP-12093, Supplement 2, "Additional Information in Support of the Evaluation of Thermal Stratification for the Beaver Valley Unit 2 Pressurizer Surge Line," August 1989.

5. WCAP-12093, Supplement 3, "Evaluation of Pressurizer Surge Line Transients Exceeding 320°F for Beaver Valley Unit 2," July 1990.

6. ASME Boiler and Pressure Vessel Code, Section III, 1986 Edition, American Society of Mechanical Engineers, New York, New York.

| Table 4.5.1-1<br>Pressurizer Surge Line Stratification Stress Analysis Summary | | | | | | |
|---|---|---|---|---|---|---|
| | Maximum Equation 12 Pipe Stress (KSI) | | Maximum Equation 13 Pipe Stress (KSI) | | Maximum Cumulative Usage Factor | |
| **Controlling Location** | **Actual Stress** | **Allowable Stress** | **Actual Stress** | **Allowable Stress** | **Actual** | **Allowable** |
| BVPS-1 Reactor Coolant Loop Nozzle | [ ]a,c | 53.0 | [ ]a,c | 50.1 | [ ]a,c | 1.0 |
| BVPS-2 Reactor Coolant Loop Nozzle | [ ]a,c | 55.5 | [ ]a,c | 48.6 | [ ]a,c | 1.0 |

## 4.6 REACTOR COOLANT PUMPS AND MOTORS

The EPU Project included analyses and evaluations for the reactor coolant pumps and motors. These analyses and evaluations are addressed in this section.

### 4.6.1 Reactor Coolant Pump Structural Integrity

#### 4.6.1.1 Introduction

This section addresses the ASME Code structural considerations for the Model 93A reactor coolant pump (RCP) pressure boundary components. The RCPs were evaluated for the EPU Project PCWG parameters and associated NSSS design transients.

This evaluation provides verification of continued structural suitability of the pressure boundary components of the existing 93A RCPs for the EPU Project.

#### 4.6.1.2 Input Parameters and Assumptions

The BVPS-1 and BVPS-2 Model 93A RCP limiting pressure boundary components were designed and analyzed to the generic component design reports and the ASME Code (See Tables 4.6.1-2A and 4.6.1-2B).

The RCPs are installed in the Reactor Coolant System (RCS) cold legs and are affected by the Reactor Coolant Pressure, Steam Generator Outlet Temperature and primary side cold leg NSSS design transients.

The input parameters that were used to perform the analyses and evaluations for EPU include the original NSSS design parameters and NSSS design transients, the EPU PCWG parameters (Section 2.1.1) along with the EPU NSSS design transients (Section 2.2.1), and the current design basis evaluations for the RCPs. It was confirmed that the original impellers remain in place on all RCPs.

#### 4.6.1.3 Description of Analyses and Evaluations

The EPU PCWG parameters and NSSS design transients are compared to the design inputs of each component as defined by the component's original evaluation for BVPS-1 and the Pressure Boundary Summary Report for BVPS-2. The results and conclusions of the original analyses are bounding and applicable provided that the input parameters used in the original analyses are shown to envelope the EPU conditions. The EPU parameters that are not bounded by the original or previous evaluations were evaluated for acceptability.

**PCWG Parameters**

The RCPs are exposed to fluid from the SG outlet nozzle. However, the temperature of the fluid at the Vessel Inlet includes the heat input from the pump. Therefore, the only PCWG parameters considered in the RCP evaluation are the Vessel Inlet Temperature and Reactor Coolant Pressure, which are evaluated for each component of the pump.

The steady state temperature distributions were determined based on an operating temperature of each pump component. Table 4.6.1-1 provides a comparison of the specified operating temperature and the corresponding design temperature for the limiting RCP pressure boundary components.

The new operating temperature is taken as the Vessel Inlet Temperature from Section 2.1.1 as 543.1°F versus the original operating temperature of 542.3°F. The minor increase in operating temperature will not impact the design temperatures of any of the pressure boundary components.

The operating and design pressures are not affected by the EPU conditions.

**Non-Enveloped Transient Evaluation Methodology**

Heat transfer evaluations were performed for non-enveloped NSSS design transients to determine the stresses due to the new temperature differential by applying a ratio of the new conditions to those conditions previously evaluated. The ratio of the new temperature differential to the original temperature differential was used to determine the stresses at EPU conditions. Any transients not previously evaluated were included in the evaluations.

#### 4.6.1.4 Acceptance Criteria and Results

The RCPs are acceptable if the design inputs from the previous analyses remain bounding and applicable to the design inputs developed for the EPU. For BVPS-1, those design inputs not enveloped were evaluated per the ASME Code editions and addenda used for the pressure boundary components in accordance with each pump component generic stress report. For BVPS-2, those design inputs not enveloped were evaluated per the ASME Code Section III edition and addenda used for the pressure boundary components in accordance with the Pressure Boundary Summary Report. Table 4.6.1-2A and Table 4.6.1-2B provide a summary of the stress intensity and fatigue usage factors for the limiting RCP pressure boundary components at EPU conditions for BVPS-1 and BVPS-2, respectively.

#### 4.6.1.5 Conclusions

The BVPS-1 and BVPS-2 RCPs were evaluated for the EPU PCWG parameters (Section 2.1.1) and the associated NSSS design transients (Section 2.2.1). In most cases, the existing analyses and evaluations remained applicable and bounding. Where this was not the case, new calculations were performed for the limiting components and the results were evaluated to establish the structural acceptability of the RCP pressure boundary components in accordance with the ASME Code.

Based on the previous analyses and the analyses and evaluations performed for EPU, the BVPS-1 and BVPS-2 RCP pressure boundary components are acceptable in accordance with the ASME Code for the EPU Project.

The results and conclusions of the analyses and evaluations performed for reactor coolant pump structural integrity for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

| Table 4.6.1-1<br>Original Operating and Corresponding Design Temperatures for Limiting RCP Pressure Boundary Components | | |
|---|---|---|
| **RCP Component** | **Operating Temperature (°F)** | **Design Temperature (°F)** |
| Main Flange Bolts | 542.3 | 600 |
| Discharge Nozzle | 542.3 | 650 |
| Suction Nozzle | 542.3 | 650 |
| Casing | 542.3 | 650 |

| Table 4.6.1-2A<br>BVPS-1 Summary of Stress Intensities and Fatigue Usage Factors for Limiting RCP Pressure Boundary Components | | | | | |
|---|---|---|---|---|---|
| **RCP Component** | **Max. Normal Stress Intensity (psi)** | **Max. Upset Stress Intensity (psi)** | **Allowable Stress Intensity (psi)** | **Max. Usage Factor** | **ASME Code(1)** |
| Main Flange Bolts | [ ]a,c | N/A(2) | 109,200 | [ ]a,c | 1968 w/Addenda through Winter 1970 |
| Discharge Nozzle | N/A(2) | [ ]a,c | 45,950 | Fatigue Waiver(3) | 1971 w/Addenda through Winter 1972 |
| Suction Nozzle | N/A(2) | [ ]a,c | 45,950 | Fatigue Waiver(3) | 1971 w/Addenda through Winter 1972 |
| Casing | N/A(2) | [ ]a,c | 45,950 | Fatigue Waiver(3) | 1971 w/Addenda through Winter 1972 |

Notes:
(1) ASME Code used in the original analysis for RCP component.
(2) This condition is not limiting.
(3) Fatigue analysis is not required per ASME III, Subsection NB since the six conditions required for exemption from fatigue analysis were satisfied.

| Table 4.6.1-2B<br>BVPS-2 Summary of Stress Intensities and Fatigue Usage Factors for Limiting RCP Pressure Boundary Components | | | | | |
|---|---|---|---|---|---|
| **RCP Component** | **Max. Normal Stress Intensity (psi)** | **Max. Upset Stress Intensity (psi)** | **Allowable Stress Intensity (psi)** | **Max. Usage Factor** | **ASME Code(1)** |
| Main Flange Bolts | [ ]a,c | N/A(2) | 109,200 | [ ]a,c | 1968 w/Addenda through Winter 1970 |
| Discharge Nozzle | [ ]a,c | [ ]a,c | 47,760 | Fatigue Waiver(3) | 1974 w/Addenda through Summer 1975 |
| Suction Nozzle | [ ]a,c | [ ]a,c | 47,760 | Fatigue Waiver(3) | 1974 w/Addenda through Summer 1975 |
| Casing | [ ]a, c (4) | [ ]a, c (4) | 47,760 | Fatigue Waiver(3) | 1974 w/Addenda through Summer 1975 |

Notes:
(1) ASME Code used in the original analysis for RCP component.
(2) This condition is not limiting.
(3) Fatigue analysis is not required per ASME III, Subsection NB since the six conditions required for exemption from fatigue analysis were satisfied.
(4) These values that exceed the $3S_m$ limit on the range of primary plus secondary stress intensity are justified by simplified elastic-plastic analysis in accordance with Paragraph NB-3228.3 in ASME III, Subsection NB.

## 4.7 STEAM GENERATORS

### 4.7.2 BVPS-2 Original Steam Generators

For BVPS-2, the Model 51M original steam generators (OSGs) were analyzed at the EPU conditions in the areas of thermal-hydraulic performance, structural integrity, U-bend fatigue, hardware changes and additions (repair hardware), tube wear, tube repair limit, and tube degradation. This section presents detailed discussions regarding the evaluations and the conclusions reached for each aspect of steam generator operation at the EPU conditions.

#### 4.7.2.1 Thermal-Hydraulic Performance

##### 4.7.2.1.1 Introduction

The original steam generators have been evaluated at the EPU conditions for 2910 MWt (970 MWt/loop) NSSS power. The thermal-hydraulic analyses for the steam generators evaluated multiple operating points at the EPU power level as well as the original BVPS-2 power level of 2660 MWt (886.7 MWt/loop) NSSS power. The EPU thermal-hydraulic evaluation considered both normal and reduced feedwater temperatures. The evaluation also considered a maximum of 22% of the steam generator tubes being removed from service by plugging (tube plugging). There are four defined operating cases for EPU operation as shown on Table 4.7.2.1-1. The thermal-hydraulic evaluation addressed eight conditions, including a normal feedwater temperature condition (455°F) and a low feedwater temperature condition (400°F) for each of the defined operating cases.

Applicable design parameters for operation at the original and the EPU power levels were used for the thermal-hydraulic evaluation. As defined in Section 2.1.1 of this report, the normal feedwater temperature is 455°F and the low feedwater temperature is 400°F. The original operating feedwater temperature for normal operation is 437.5°F at 100% power. Section 2.1.1 of this report defines the high and low coolant temperatures at the steam generator inlet as 617° and 603.9°F, respectively, for operation at the EPU conditions. This temperature is 609.9°F for operating at the original power level. The operating steam generator water level was set at the normal level, 514 inches above the tube sheet surface. The fouling factor was taken at the design value for all conditions, which are 0.000160 hr-ft$^2$-°F/Btu for the original power and 0.000055 hr-ft$^2$-°F/Btu for the EPU.

#### 4.7.2.1.2 Input Parameters and Assumptions

The eight conditions (two feedwater temperatures for each of the four defined operating cases) used for the steam generator thermal-hydraulic evaluations for the EPU are defined in Section 2.1.1 of this report and are shown on Table 4.7.2.1-1. The thermal-hydraulic evaluation of the steam generators focused on secondary side operating characteristics.

#### 4.7.2.1.3 Description of Analyses and Evaluations

##### 4.7.2.1.3.1 Thermal-Hydraulic Operating Characteristics

Secondary-side steam generator performance characteristics may be affected by changes in steam flow and steam pressure. The evaluations are performed for EPU conditions. The primary temperature and steam generator tube plugging (SGTP) level affect steam pressure. Steam flow increases and steam pressure decreases when power is increased. In addition, steam flow at constant power decreases as feedwater temperature decreases. This section assesses the magnitude and importance of changes in the secondary-side thermal-hydraulic performance characteristics due to EPU at eight design conditions (normal and reduced feedwater temperatures for each of the four defined operating cases), as defined in Section 2.1.1, NSSS Design (PCWG) Parameters, of this report and shown in Table 4.7.2.1-1.

The EPU results in an increase in steam flow by an amount of about 12% for the normal feedwater temperature (455°F) and about 4% for the reduced feedwater temperature (400°F). The increase of 12% in steam flow is expected due to the combined effect of a power increase to 2910 MWt and a feedwater temperature increase from 437.5° to 455°F. The increase of 4% in steam flow is expected due to the combined effect of a power increase to 2910 MWt and a feedwater temperature decrease from 437.5° to 400°F.

Most of the operating characteristics displayed expected small or benign changes as a result of a change in the feedwater temperature. Steam flows decrease for the reduced feedwater temperatures as a result of the increased enthalpy difference at constant power. The steam pressures are not affected by feedwater temperature. Circulation ratio increases proportionally to the change in steam flow as is typical for re-circulating steam generators.

At a given power, average heat fluxes are proportional to the heat transfer area in service and are not affected by the feedwater temperature. Peak heat fluxes are similarly unaffected by feedwater temperature. Steam generator secondary liquid mass increases with reduced feedwater temperature; the increase is up to 2% over the original power. Steam generator secondary liquid mass decreases with normal feedwater temperature; the decrease is up to 7% less than at the original power.

Contaminant that enters with feedwater appears as particulate, colloids or dissolved chemical; it can deposit on the tube surface or settle on the tubesheet. The amount of the contaminant is proportional to the feedwater flow rate and the contaminant concentration. For a constant contaminant concentration, the total quantity of contaminant on the tubesheet and tube bundle would potentially increase by 12% for normal feedwater temperature (455°F) and 4% for reduced feedwater temperature (400°F) where the assumption of equality of feedwater flow and steam flow is considered. Of course, contaminant concentration is hardly constant and can vary significantly for a constant power.

#### 4.7.2.1.3.2 Moisture Carryover

Moisture separator performance depends on steam flow, steam pressure, and water level. The EPU maintains the same secondary-side water level as that for the original power. The amount of moisture carryover (MCO) tends to increase as steam flow increases or as steam pressure decreases. The EPU results in an increase in steam flow as well as a decrease in steam pressure. Therefore, the amount of moisture carryover is higher for the EPU than for the original power. However, the EPU will not experience excessive moisture carryover. The moisture carryover ranges from [ ]$^{a,c}$ % to [ ]$^{a,c}$ % for BVPS-2. The MCO will remain less than the design limit of 0.25%.

#### 4.7.2.1.3.3 Hydrodynamic Stability

If hydrodynamic stability is not maintained, all of the thermal and hydraulic parameters will oscillate. For example, excessive oscillations in water level can then take place and result in control problems. The hydrodynamic stability of a steam generator is characterized by the damping factor. A negative value of this parameter indicates a stable unit, and thus a stable water level. That is, small perturbations of steam flow or pressure will die out rather than grow in amplitude. A number of factors can have a destabilizing effect on steam generators. These factors include: (1) power level, (2) increased downcomer sub-cooling, and (3) steam pressure. For BVPS-2, reduced stability could be caused by: (1) increased power, (2) decreased steam pressure resulting from power increase, and (3) increased downcomer sub-cooling, resulting from the cases with reduced feedwater temperature added to the downcomer. However, there are cases with normal feedwater temperature that lead to increased stability when compared to the original power.

For the reduced feedwater temperature cases, the damping factor ranges from [ ]$^{a,c}$ $hr^{-1}$ to [ ]$^{a,c}$ $hr^{-1}$ for BVPS-2. For the normal feedwater temperature cases, the damping factor ranges from [ ]$^{a,c}$ $hr^{-1}$ to [ ]$^{a,c}$ $hr^{-1}$ for BVPS-2. Note that the original power results in a damping factor of [ ]$^{a,c}$ $hr^{-1}$ for BVPS-2. The damping factors at reduced feedwater temperature, however, remain substantially negative. Therefore the steam generators will continue to be hydrodynamically stable.

#### 4.7.2.1.3.4 Local Dryout of Tubes

Local dryout on the tube wall is also called departure from nucleate boiling (DNB). The DNB results from liquid deficiency on the tube wall, and thus can trigger the excessive build-up of tube scale. The excessive tube scale may result in chemical concentration at that locality and thus possible tube corrosion. The DNB is a localized phenomenon due to several factors, such as circulation ratio, steam pressure, heat flux, and steam flow rate. Since the steam generator operates at a power higher than the original 100% rated power, the circulation ratio decreases, steam pressure also decreases, and heat flux increases, and these tendencies will increase the potential of DNB.

Evaluations of all relevant parameters were made for the EPU conditions. As expected, potential of DNB on the tubes increases towards the upper bundle and is highest at the U-bend. However, the evaluation revealed that even the U-bend has approximately 40% margin to the DNB. Therefore, it is concluded that there is no concern of local dryout on the tube walls for operation at the EPU conditions.

#### 4.7.2.1.4 Acceptance Criteria and Results

Several secondary-side operating characteristics were used to assess the acceptability of steam generator operation at EPU conditions. These parameters included steam flow and pressure; circulation ratio; damping factor, which is a measure of hydrodynamic stability; local (liquid) dryout on tube wall, which addresses the combined effect of secondary mass flow; heat flux; and secondary side pressure. Moisture carryover is also affected by the EPU and is addressed. All of these parameters were evaluated with respect to two issues: whether they create operational difficulty, and whether they compromise component integrity. With respect to these two issues, acceptable steam generator performance is demonstrated by: (1) no excessive moisture carryover, (2) no hydrodynamic instability, and (3) no local dryout on tube walls. As demonstrated in Section 4.7.2.1.3, all acceptance criteria continue to be met.

#### 4.7.2.1.5 Conclusions

All projected thermal-hydraulic operating characteristics for the four EPU operating cases are acceptable for both the 400° and 455°F feedwater temperatures. There are no concerns of thermal performance deficiency, excessive moisture carryover, hydrodynamic instability or local dryout on tube walls.

The results and conclusions of the analyses and evaluations performed for steam generator thermal hydraulic performance for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

| Table 4.7.2.1-1<br>BVPS-2 Results of Thermal Hydraulic Evaluations | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| **Case #** | **0** | **1a** | **1b** | **2a** | **2b** | **3a** | **3b** | **4a** | **4b** |
| **Operating Conditions as Defined by Section 2.1.1** | | | | | | | | | |
| Power, % | 100 | 109.4 | 109.4 | 109.4 | 109.4 | 109.4 | 109.4 | 109.4 | 109.4 |
| NSSS Power, MWt | 2660 | 2910 | 2910 | 2910 | 2910 | 2910 | 2910 | 2910 | 2910 |
| Power, MWt/SG | 886.67 | 970 | 970 | 970 | 970 | 970 | 970 | 970 | 970 |
| Primary Flow, gpm/loop | 88500 | 87200 | 87200 | 87200 | 87200 | 87200 | 87200 | 87200 | 87200 |
| $T_{hot}$, °F | 609.9 | 603.9 | 603.9 | 603.9 | 603.9 | 617.0 | 617.0 | 617.0 | 617.0 |
| $T_{cold}$, °F | 542.5 | 528.5 | 528.5 | 528.5 | 528.5 | 543.1 | 543.1 | 543.1 | 543.1 |
| $T_{avg}$, °F | 576.2 | 566.2 | 566.2 | 566.2 | 566.2 | 580.0 | 580.0 | 580.0 | 580.0 |
| Feed Temp, °F | 437.5 | 400 | 455 | 400 | 455 | 400 | 455 | 400 | 455 |
| Plugging, % | 0 | 0 | 0 | 22 | 22 | 0 | 0 | 22 | 22 |
| Fouling, hr-ft$^2$-°F/Btu x $10^{-6}$ | 160 | 55 | 55 | 55 | 55 | 55 | 55 | 55 | 55 |
| **T/H Characteristics of Steam Generator** | | | | | | | | | a,c |
| Steam Flow, $Mlb_m$/hr per SG | | | | | | | | | |
| Steam Pressure, psia | | | | | | | | | |
| Steam Temperature, °F | | | | | | | | | |
| Circulation Ratio, CR | | | | | | | | | |
| Separator Parameter | | | | | | | | | |
| Total 2$^{nd}$ Pressure Drop, psi | | | | | | | | | |
| Downcomer Velocity, ft/sec | | | | | | | | | |
| Downcomer Temp., °F | | | | | | | | | |
| 2$^{nd}$ Side Liquid Mass, $lb_m$ | | | | | | | | | |
| 2$^{nd}$ Side Vapor Mass, $lb_m$ | | | | | | | | | |
| 2$^{nd}$ Side Fluid Heat, Million Btu | | | | | | | | | |
| Average heat flux, Btu/hr-ft$^2$ | | | | | | | | | |
| Damping Factor, $hr^{-1}$ | | | | | | | | | |
| U-bend Fluid Velocity, ft/sec | | | | | | | | | |
| U-bend Fluid Density, $lb_m/ft^3$ | | | | | | | | | |
| U-bend Quality(1/CR) | | | | | | | | | |
| U-bend Void Fraction | | | | | | | | | |
| **Moisture Carryover – Average for all Generators** | | | | | | | | | |
| MCO, % | [ | | | | | | | | ]$^{a,c}$ |

### 4.7.2.2 Structural Integrity

#### 4.7.2.2.1 Introduction

An evaluation was performed to assess the structural integrity of the original steam generators during operation at the EPU power level with steam generator tube plugging (SGTP) in the range of 0% to 22%. The NSSS Design Transients in Section 2.2.1 were used to generate scaling factors with respect to the original stress reports results. The scaling factors were based on the Low $T_{avg}$, 22% plugging (Case 2) condition, limited to a minimum steam pressure of 700 psia to meet the design specification $\Delta P$ requirement of 1600 psi.

Additional evaluations were performed to address the effects of the Cold Overpressure Mitigation System (COMS) and $T_{avg}$ Coastdown.

#### 4.7.2.2.2 Input Parameters and Assumptions

The PCWG parameters used for the steam generator's structural evaluation are provided in Section 2.1.1. For the primary side components, the enveloping condition resulted in the largest pressure differential between the primary and secondary sides of the steam generator among the PCWG cases.

For the secondary side components, operation in the EPU condition results in reduction in the steam pressure, thereby increasing the primary-to-secondary pressure differential. The resulting additional pressure stresses are added to the stresses reported in the original stress report to calculate the revised stress range and revised fatigue usage.

#### 4.7.2.2.3 Description of Analyses and Evaluations

Scaling factors were calculated based on the ratio between the primary-to-secondary differential pressure in the EPU condition to that of the reference stress report condition. The scale factors are applied to the stress range and alternating stresses taken from the generic Model 51 stress reports to estimate the revised stress range and revised fatigue usage for operation in the EPU condition.

The PCWG parameters for Low $T_{avg}$, with 22% SGTP show the steam pressure ($P_{stm}$) to be 641 psia. With a steam pressure of 641 psia, the primary-to-secondary $\Delta P$ for the Normal operation condition, considering design transients, would be 1641 psi. This would violate the maximum 1600 psi $\Delta P$ requirements of the design specification. Limiting $P_{stm}$ for Low $T_{avg}$ operation to 700 psia for the 22% plugging case would result in a primary-to-secondary side $\Delta P$ for the Normal operating condition of 1588 psi. This pressure limit satisfies the design specification requirements of 1600 psi.

On this basis the evaluations for EPU were based on the $P_{stm}$ of 700 psia. The scaling factors were calculated with reference to the stress report case of $P_{stm}$ = 790 psia.

**Primary Side Components**

The stress in the primary side components is primarily dependent on the differential pressure between the primary side and secondary side. The stress in the secondary side components is primarily dependent on the steam pressure. The increase of pressure stress due to a reduction in steam pressure as a result of the EPU was considered in calculating the increase in the stress range and the resulting increase in the fatigue usage.

The scale factors were calculated for the EPU with reference to the stress report 100% power conditions for each of the Normal/Upset transients that were addressed in the original stress report. As previously noted, since the design specification limits the primary-to-secondary side ΔP to ≤1600 psi, the scaling factors were based on a steam pressure of 700 psia at 100% EPU power.

The scale factor at 100% EPU power was calculated as shown below:

$$\frac{\Delta P_{EPU}}{\Delta P_{original}} = \frac{(2250 - 700)_{EPU}}{(2250 - 790)_{original}} = 1.062$$

The scaling factors were applied to the transients involved in the stress range and fatigue evaluation reported in the original stress report. The revised stress range and fatigue usage were calculated for primary side components using this scale factor.

**Secondary Side Components**

The stress in the secondary side components is primarily dependent on the steam pressure. The reduction in steam pressure due to the EPU results in an increase in the primary-to-secondary ΔP. This results in an increase in the secondary side component stresses. To estimate the component stresses in the EPU conditions, it is possible to increase the reference stress report stresses by the incremental stress that would result from the change in pressure.

The secondary side component stresses were scaled as shown below:

> Stress range $_{(EPU)}$ = [Stress range $_{(stress\ report)}$ + pressure stress due to 90 psi],
>
> Where 90 psi is the difference in steam pressure from the Stress Report Case to the EPU Case, or, [790 psia ($P_{stm\ (Stress\ Report)}$) - 700 psia ($P_{stm\ (EPU)}$)].

**COMS Transient Evaluation**

The COMS transient conservatively assumes that maximum and minimum RCS pressures fluctuate from 195 psia to 800 psia over a period of 1 second. Since the span of the COMS event is 10 minutes, and there are 60 cycles per minute, the total number of cycles is 600 per event, or 6000 cycles during an operation period. The COMS transient is considered to be an Upset condition.

The COMS pressure cycles are similar to primary side pressure tests. The stresses due to the COMS transient are calculated by scaling the stresses from the Primary Hydro Test or Primary Leak Test transient.

The stress reports for the original steam generators did not consider the effect of the COMS transients. However, for BVPS-2, a supplemental evaluation had been performed to address the effects of COMS. This evaluation only calculated revised fatigue usage values for the most critical components. Therefore, for the EPU evaluations, the stress analysis for BVPS-2 was revised to include a more in-depth evaluation of the effect of the COMS transients. Only the primary side components are affected by the COMS transients.

**$T_{avg}$ Coastdown Transient Evaluation**

For this evaluation, the $T_{avg}$ Coastdown event is considered to occur at the end of each operational cycle until the expiration of the current 40-year operating license.

Since the Coastdown maneuver occurs at 100% power loading conditions, the Coastdown evaluation was done for 100% power loading condition, based on a minimum steam pressure of 700 psia.

The number of coastdown cycles and their impact were determined as follows:

1. It is assumed that the $T_{avg}$ Coastdown occurs at the end of each remaining operational fuel cycle until the expiration of the current 40-year operating license. The length of the fuel cycle is 18 months.

2. Twenty (20) Coastdown cycles are considered in the evaluation.

3. The fatigue usage due to the $T_{avg}$ Coastdown transient is added to the fatigue usage that has been evaluated due to the EPU only.

The number of $T_{avg}$ Coastdown cycles remaining until the end of the design period was calculated to be 17 cycles for BVPS-2. Conservatively, 20 Coastdown cycles are used to evaluate fatigue usage due to Coastdown.

The stress and fatigue calculations were scaled from the generic stress reports. The stress and fatigue of the critical structural components were evaluated considering 20 Coastdown transients. The Coastdown fatigue was added to the fatigue evaluation done for the EPU, and the total fatigue usage considers both the EPU and Coastdown events.

**Primary-to-Secondary Pressure Differential (Delta P) Evaluation**

An analysis was performed to determine the minimum acceptable full power steam pressure for BVPS-2 for the EPU condition. The design pressure limit for primary-to-secondary pressure differential is 1600 psi as defined in the applicable design specifications. The design pressure requirement for Class 1 Equipment is defined in the ASME Code, Section III. The applicable edition of the Code for the BVPS-2 steam generators is 1971, Summer 1972 Addenda.

As a result of the Code review, it is concluded that the Normal/Upset transient conditions are subject to the following design pressure requirements.

1. Normal Condition Transients: Primary-to-Secondary pressure gradient shall be less than the design limit of 1600 psi.

2. Upset Condition Transients: If the pressure during an upset transient exceeds the design pressure limit, the stress limits corresponding to design conditions apply using an allowable stress intensity value of 110% of those defined for Design Conditions. In other words, so long as the Upset Condition pressure values are less than 110% of the design pressure values, no additional analysis is necessary. For the BVPS-2 steam generators, 110% of the design pressure limit corresponds to 1760 psi.

The analysis is based on the transient parameters in Section 2.2.1. Two sets of transient parameters are defined, one corresponding to a High $T_{avg}$ mode of operation and one corresponding to a Low $T_{avg}$ mode of operation. In addition, transient parameters are defined for two different tube plugging levels, 0% and 22%. The pressure differentials across the primary-to-secondary side pressure boundary are calculated for all four sets of conditions.

A summary of the analysis results is provided in Table 4.7.2.2-1. The analysis results show that, with the exception of Case 2, the maximum primary-to-secondary pressure gradients for Cases 1 through 4 are less than the allowable values. For the Low $T_{avg}$, 22% plugging (Case 2) condition, the maximum primary-to-secondary pressure gradient for the normal transient is 1641 psi (the limiting transient is the 10% Load Increase transient).

Thus, calculations were performed to determine the minimum full power steam pressure that is required such that the maximum primary-to-secondary pressure gradient is ≤1600 psi. The results show that for a minimum full power steam pressure of 687 psia ($T_{steam}$ = 501.0°F), the maximum primary-to-secondary pressure gradient for the Normal Condition case limiting 10% Load Increase transient is 1599 psi. The corresponding maximum primary-to-secondary pressure gradient for the Upset Condition case limiting Loss of Flow transient is 1578 psi, which is less than the allowable value of 1760 psi. Thus, the minimum acceptable full power steam pressure is 687 psia. In order to provide some conservatism relative to the design pressure limit, plant operation is recommended to be maintained at or above a full power steam pressure of 700 psia ($T_{steam}$ = 503.1°F). The allowable steam pressure of 687 psia and the recommended steam pressure of 700 psia cases are included in Table 4.7.2.2-1 as Case 5 and Case 6, respectively.

**Secondary Manway Bolt Fatigue Evaluation**

The fatigue usage for the secondary manway bolts was calculated based on the minimum secondary side steam pressure of 760 psia for operation prior to the EPU, and 700 psia for operation after the EPU. The cumulative fatigue usage for 40 years was calculated by combining the fatigue usages for the pre-EPU and post-EPU operation.

The fatigue usage for 40 years of operation was calculated for both 760 psia and 700 psia operation. To determine the cumulative fatigue usage for operation at the given pressure for less than the full design life (40 years), the effective usage was determined by multiplying the 40 year values by a ratio of actual years over design life.

@ 760 psia for 40 years, usage = [ ]$^{a,c}$

@ 700 psia for 40 years, usage = [ ]$^{a,c}$

The steam generators were considered to operate at 760 psia from August of 1987 to June of 2002 (Approximately 15 years), and at 700 psia for the remaining 25 years of the design life. The effective usage for operation at the two different pressure levels was then;

[ ]$^{a,c}$ (15/40) + [ ]$^{a,c}$ (25/40) = [ ]$^{a,c}$

It was then necessary to add the additional fatigue usage that occurs due to the planned $T_{avg}$ coastdown maneuvers. The fatigue usage due to the coastdowns was determined in the $T_{avg}$ coastdown transient evaluation to be [ ]$^{a,c}$. The total cumulative fatigue usage for the bolts, over a 40 year design life, was then determined to be:

[ ]$^{a,c}$ < 1.0

This demonstrated that the secondary manway bolts are adequate for the 40 year design life of the plant, even after the EPU is implemented.

#### 4.7.2.2.4 Acceptance Criteria and Results

The acceptance criteria for each component is consistent with the criteria used in the design basis analysis referenced for that component. The maximum range of primary-plus-secondary stresses were compared with the corresponding $3S_m$ limits, Reference 1. For situations where these limits were exceeded, a simplified elastic-plastic analysis was performed per NB 3228.3 of Reference 1. A cumulative fatigue usage factor below unity demonstrates the adequacy for a 40-year design life.

The critical components considered in the EPU were evaluated for the: 1) EPU condition, 2) COMS transient, and 3) Coastdown transient. The results of the evaluation show that all components analyzed meet ASME code limits. The results of the evaluation are summarized in Table 4.7.2.2-2.

#### 4.7.2.2.5 Conclusions

Results of the analyses performed on the steam generators show that the ASME Code Section III limits are met for the critical structural components for operation at the EPU condition.

The results and conclusions of the analyses and evaluations performed for steam generator structural integrity for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

#### 4.7.2.2.6 References


1. ASME Boiler and Pressure Vessel Code, Section III, "Rules for the Construction of Nuclear Vessels," 1971 Edition, plus Addenda through Summer 1972, ASME, New York, New York.

| Table 4.7.2.2-1<br>**BVPS-2 Summary of Delta P Evaluation for Operation at the EPU Condition** | | | | | |
|---|---|---|---|---|---|
| **Case** | **Item** | **Limiting Transient** | **Case** | **Delta P psi** a,c | **Allow. psi** |
| 1 | Low $T_{avg}$ 0% Plugging | 10% Load Increase | Normal | | 1600 |
| | | Loss of Flow | Upset | | 1760 |
| 2 | Low $T_{avg}$ 22% Plugging | 10% Load Increase | Normal | | 1600 |
| | | Loss of Flow | Upset | | 1760 |
| 3 | High $T_{avg}$ 0% Plugging | 10% Load Increase | Normal | | 1600 |
| | | Loss of Flow | Upset | | 1760 |
| 4 | High $T_{avg}$ 22% Plugging | 10% Load Increase | Normal | | 1600 |
| | | Loss of Flow | Upset | | 1760 |
| 5 | Low $T_{avg}$ 22% Plugging | 10% Load Increase | Normal | | 1600 |
| | (For Minimum Allowable $P_{stm}$ = 687 psia) | Loss of Flow | Upset | | 1760 |
| 6 | Low $T_{avg}$ 22% Plugging | 10% Load Increase | Normal | | 1600 |
| | (Recommended $P_{stm}$ = 700 psia) | Loss of Flow | Upset | | 1760 |

**Table 4.7.2.2-2**
**BVPS-2 EPU Structural Integrity Evaluation Summary**

| Component | Load Condition | Stress Category | Original Cond. Stress (ksi)/ Fatigue | EPU Cond. Stress (ksi)/ Fatigue | COMS Stress (ksi)/ Fatigue(6) | Coastdown Stress (ksi)/ Fatigue | Total Fatigue | Allow (ksi)/ Fatigue | Comments |
|---|---|---|---|---|---|---|---|---|---|
| Divider Plate | Normal/upset | $P_m+P_b+Q$ | comment | comment | [ ]a,c | [ ]a,c | n/a | 69.90 | Elastic-Plastic Analysis(11) |
| | | Fatigue(9) | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | 1.00 | |
| Tubesheet & Shell Junction | Normal/upset | $P_m+P_b+Q$ | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | n/a | 80.10 | |
| | | Fatigue | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | 1.00 | |
| Tube to Tubesheet Weld | | $P_m+P_b+Q$ | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | n/a | 69.90 | |
| | Normal/upset | Fatigue | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | [ ]a,c | 1.00 | |
| Tubes | Design (St leg) | $S_m$ | [ ]a,c | [ ]a,c | notes(7) | notes(7) | notes(7) | 23.30 | |
| | Design (U bend) | $S_m$ | [ ]a,c | [ ]a,c | notes(7) | notes(7) | notes(7) | 23.30 | |
| Main Feed Water Nozzle | Normal/upset | $P_m+P_b+Q$(1) | [ ]a,c | [ ]a,c | n/a | [ ]a,c | n/a | 80.10 | |
| | | Fatigue(5) | [ ]a,c (5) | [ ]a,c (4,5) | n/a | [ ]a,c | [ ]a,c | 1.00 | |
| Secondary Manway Bolts | Normal/upset | $P_m+P_b+Q$(2) | [ ]a,c | [ ]a,c | n/a | [ ]a,c | n/a | 99.00 | |
| | | Fatigue(8) | [ ]a,c | [ ]a,c (10) | n/a | [ ]a,c | [ ]a,c (10) | 1.00 | |
| Steam Nozzle | Normal/upset | $P_m+P_b+Q$ | [ ]a,c | [ ]a,c | n/a | [ ]a,c | n/a | 80.10 | |
| | | Fatigue(3) | [ ]a,c | [ ]a,c | n/a | [ ]a,c | [ ]a,c | 1.00 | |

**Table 4.7.2.2-2 (continued)**
**BVPS-2 EPU Structural Integrity Evaluation Summary**

Notes:

(1) Additional stress due to reduction of steam pressure is taken to calculate the increase in stress range for secondary side components.

(2) Change in steam pressure has no effect since the maximum stress range occurs between the hydrotest and bolt installation.

(3) Based on OBE = 400 cycles.

(4) The conservative lumping of the transients used in the stress report is revised in the EPU evaluation.

(5) Fatigue is calculated for OBE = 400 cycles; Note fatigue values were reduced by changing the conservative combination of transients of the original analysis.

(6) COMS are applicable to Primary side components only.

(7) Design condition is not affected due to EPU. The U-bend fatigue and Reg. Guide 1.121 Evaluations are done separately due to EPU.

(8) The fatigue usage is calculated based on the minimum secondary side pressure of 760 psia for operation prior to EPU, and 700 psia for operation after EPU.

(9) The fatigue usage is calculated using conservative scale factors, based on the maximum number of cycles specified in the E-Spec. This is a conservative approach.

(10) Fatigue usage factor for BVPS-2 secondary manway bolts.

(11) Values that exceed the $3S_m$ limit on the range of primary plus secondary stress intensity are justified by simplified elastic-plastic analysis in accordance with Paragraph NB-3228.3 in ASME III, Subsection NB.

### 4.7.2.3 U-bend Fatigue

#### 4.7.2.3.1 Introduction

Changes in secondary side conditions as a result of the EPU can affect the previously completed evaluation performed to address the effects of high cycle fatigue in unsupported steam generator U-bend tubes. The following addresses the effects of the EPU on the previously completed U-bend fatigue analysis. For any tubes found to have high fatigue usage factors after the EPU, appropriate recommendations are provided below.

#### 4.7.2.3.2 Input Parameters and Assumptions

The following are a list of major assumptions used in preparation of this analysis:

1. BVPS-2 will continue to operate for an additional 26 years. This is considered a reasonable estimate since the operating license is currently set to expire in 2027.

2. Eight EPU conditions are considered in the analysis. Table 4.7.2.3-1 contains a summary of these conditions.

3. Tubes identified in Table 4.7.2.3-2 have already been removed from service as a result of previously completed U-bend fatigue analysis.

4. Effect of BVPS-2 venturi inaccuracies have been incorporated.

#### 4.7.2.3.3 Description of Analyses and Evaluations

The U-bend fatigue evaluations for BVPS-2 at the original power level is provided in Reference 1. The U-bend fatigue evaluation performed for EPU used the same methodology as the original evaluations.

##### 4.7.2.3.3.1 Relative Stability Ratio

Stability ratio is a ratio of fluid velocity to threshold velocity at or above which flow induced tube vibration will occur. Relative stability ratio (RSR) is a ratio of stability ratio to a reference stability ratio. The reference stability ratio is generally taken to be that of North Anna Unit 1 because that plant had experienced tube vibration and fatigue rupture. Relative stability ratio to North Anna Unit 1 (VRA) is calculated with known operating conditions of steam flow, circulation ratio and steam pressure. The relative stability ratios are calculated based on a one-dimensional analysis and are referred to as 1D-RSRs. Results are tabulated in Table 4.7.2.3-3 under the column labeled "RSR to North Anna."

The reference stability ratio can be any plant at any operating conditions. Generally, this alternative stability ratio is taken to be that at the full power operation conditions of a specific plant. The second definition is also used for this evaluation. The specific plant for the current evaluation is BVPS-2. Results of relative stability ratio to the BVPS-2 full power conditions can be readily calculated from "RSR to North Anna." This has been performed with results tabulated for BVPS-2 in Table 4.7.2.3-3 under the column labeled "RSR to BVPS-2-Ref."

#### 4.7.2.3.3.2 Fatigue Usage

The effects of the EPU operating conditions on unsupported U-bend tubes have been determined by calculating the total fatigue usage for the most limiting potentially susceptible tubes that have not been previously identified for preventative action. A list of these potentially susceptible tubes can be found in Table 4.7.2.3-4 for BVPS-2. This evaluation has been performed by modifying the previously calculated tube specific relative stability ratios using the plant specific 1D-RSRs discussed above. The revised stress ratios and subsequent fatigue usages are then calculated using these relative stability ratios. This calculation includes fatigue usage accumulated prior to the EPU and also subsequent to the EPU at the various EPU operating conditions. The BVPS-2 fatigue calculation also includes the effects of the venturi inaccuracies. Any tubes requiring preventive action have been identified for each of the various EPU operating conditions. Calculations performed in support of the current analysis are an application of previously reported methods.

### 4.7.2.3.4 Acceptance Criteria and Results

#### 4.7.2.3.4.1 Case Specific Results

Total fatigue usage has been calculated for the various EPU cases by summing the fatigue usage associated with:

1. Previous operation
2. Effects of venturi inaccuracies for BVPS-2
3. Future operation at the various EPU operating conditions, including $T_{avg}$ Coastdowns

The past and future operation conditions were used along with the 1D-RSRs to calculate a total cumulative fatigue usage for each of the enveloping tubes for each of the proposed EPU operating conditions. Tubes with fatigue usages greater than 1.0 were then recommended for appropriate preventative action such as installation of sentinel plugs or installation of cable tube dampers. Table 4.7.2.3-4 contains a summary indicating which tubes were found to have exceeded a 1.0 fatigue usage factor at the operating conditions for the various EPU cases. Note that from 0 to 6 tubes for BVPS-2 could require preventative action, depending upon the particular operating conditions for the EPU cases.

#### 4.7.2.3.4.2 Limiting RSR and Minimum Steam Pressure Calculations

Calculations were then performed to determine the maximum RSR multiplier, and the corresponding operating condition, that would be required to determine the acceptability of each of the enveloping tubes. Table 4.7.2.3-5 contains a summary of the limiting RSR values along with the associated operating condition for each of these tubes. The conditions necessary for the enveloping tubes to obtain a total fatigue usage factor of 1.0 have been identified in this table. It should be noted that two main feedwater temperature conditions of 400° and 455°F have been considered in the EPU analysis. For each of these feedwater temperature conditions, the minimum acceptable steam pressure has been identified.

A summary of these results indicating which tubes would be required to be removed from service at a given feedwater temperature and steam pressure, can also be found in Table 4.7.2.3-5.

#### 4.7.2.3.4.3 Results for Minimum Steam Pressure of 700 psia

As described in Section 4.7.2.2 and noted in Section 2.1.1, the minimum full power steam generator outlet steam pressure at EPU operating conditions is limited to 700 psia in order to satisfy the steam generator primary-to-secondary differential pressure design limit of 1600 psi. This minimum steam pressure can be used in conjunction with the information contained in Table 4.7.2.3-5 to reduce the number of tubes requiring action to remove from service for EPU operating conditions. Since steam pressure will be controlled above 700 psia during full power operation, Table 4.7.2.3-5 shows that Tube Group 4 can be removed from the list of tubes requiring action for EPU operating conditions including the 700 psia minimum steam pressure. Deletion of Tube Group 4 reduces the number of tubes requiring action by 2 tubes for BVPS-2. The remaining number of tubes requiring action is 4 for BVPS-2. A comparison of the information in Table 4.7.2.3-5 to the information in Table 4.7.2.3-4 shows that controlling steam pressure above 700 psia during full power operation eliminates PCWG parameter Case 2b as the limiting case for identification of tubes requiring action. PCWG parameter Case 1b then becomes the limiting case and identifies the tubes requiring action to support EPU for BVPS-2.

#### 4.7.2.3.5 Conclusions

Results of the analysis showed that under the most limiting EPU operating conditions, up to 6 tubes for BVPS-2 could require removal from service. These tubes and groups of tubes are identified in Table 4.7.2.3-5 as Tube Groups 1, 2, 3 and 4. However, since the primary drivers for the analysis relate to feedwater temperature and steam pressure, the analysis showed that a reduced number of tubes might require preventative action depending upon the specifics of the actual operating conditions.

Including consideration of the 700 psia minimum steam pressure in the evaluation reduced the number of tubes requiring preventative action to 4 for BVPS-2. The tubes requiring action are identified in Tube Groups 1, 2 and 3 in Table 4.7.2.3-5. They are also identified in Table 4.7.2.3-4 under Case 1b.

Tubes identified as requiring preventative action will be removed from service using sentinel plugs, or have cable tube dampers installed.

The results and conclusions of the analyses and evaluations performed for steam generator U-bend fatigue for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

#### 4.7.2.3.6 References


1. WCAP-12141, Rev. 1, "Beaver Valley Unit 2 88-02 Re-Evaluation for Tube Vibration Induced Fatigue," April 1993.

| Table 4.7.2.3-1<br>BVPS-2 EPU Case Descriptions | | | |
|---|---|---|---|
| **Case Name** | **Steam Flow 10$^6$ lb$_m$/hr** | **Circ. Ratio** | **Steam Pressure** |
| Case 0(1) | | | a,c |
| Case 1a | | | |
| Case 1b | | | |
| Case 2a | | | |
| Case 2b | | | |
| Case 3a | | | |
| Case 3b | | | |
| Case 4a | | | |
| Case 4b | | | |

Note:
(1) Case 0 – Original operating conditions

| Table 4.7.2.3-2<br>BVPS-2 List of Tubes Previously Removed from Service as a Result of Previously Completed U-Bend Fatigue Analysis | | |
|---|---|---|
| **Unit** | **SG** | **Row/Column** |
| 2 | C | R11C4 |
| 2 | C | R9C33 |
| 2 | A | R9C84 |
| 2 | C | R9C84 |
| 2 | A | R8C60 |
| 2 | C | R8C69 |
| 2 | B | R11C5(1) |

Note:
(1) This tube was previously removed from service by plugging. It requires deplugging with installation of a sentinel plug.

| Table 4.7.2.3-3<br>BVPS-2 Relative Stability Ratio | | | | | | |
|---|---|---|---|---|---|---|
| **Plant Alpha** | **Plant Name or Case No.** | **Steam Flow, $10^6$ $lb_m/hr$** | **Circulation Ratio** | **Steam Pressure, psia** | **RSR to North Anna** | **RSR to BVPS-2-Ref** |
| VRA | North Anna 1 | | | | | a,c |
| BVPS-2 | BVPS-2 Reference Case | | | | | |
| BVPS-2 | Case 0 | | | | | |
| BVPS-2 | Case 1a | | | | | |
| BVPS-2 | Case 1b | | | | | |
| BVPS-2 | Case 2a | | | | | |
| BVPS-2 | Case 2b | | | | | |
| BVPS-2 | Case 3a | | | | | |
| BVPS-2 | Case 3b | | | | | |
| BVPS-2 | Case 4a | | | | | |
| BVPS-2 | Case 4b | | | | | |

**Table 4.7.2.3-4**
**BVPS-2 Tubes Requiring Action for Each EPU Case(1,2,3)**

| Unit | S.G. | Row | Columns | No. Tubes in Group | Case Number and Case Name | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|---|---|---|---|
| | | | | | 1<br>Case 2b | 2<br>Case 1b | 3<br>Case 2a | 4<br>Case 4b | 5<br>Case 1a | 6<br>Case 3b | 7<br>Case 4a | 8<br>Case 3a | 9<br>Case 0 |
| 2 | B | 11 | 3 through 4 | 2 | Plug | Plug | Plug | Plug | Plug | ok | ok | ok | ok |
| 2 | B | 11 | 5 | 1 | Plug | Plug | Plug | Plug | ok | ok | ok | ok | ok |
| 2 | C | 9 | 42 | 1 | Plug(4) | ok | ok | ok | ok | ok | ok | ok | ok |
| 2 | C | 9 | 43 | 1 | Plug(4) | ok | ok | ok | ok | ok | ok | ok | ok |
| 2 | C | 10 | 4 | 1 | Plug | Plug | Plug | Plug | ok | ok | ok | ok | ok |
| Total No. Tubes Requiring Preventative Action = | | | | | 6 | 4 | 4 | 4 | 2 | 0 | 0 | 0 | 0 |

Notes:

(1) Case descriptions given in Table 4.7.2.3-1.

(2) Assumes tubes in Table 4.7.2.3-2 already plugged with sentinel plugs.

(3) "Plug" indicates removal from service, usually with sentinel plugs.

(4) No action required since associated steam pressure is maintained above the minimum steam pressure of 700 psia required for full power operation at EPU conditions.

**Table 4.7.2.3-5**
**BVPS-2 Enveloping Tubes and Minimum Acceptable Steam Pressures(1)**

| Tube Group No. | Total No. of Tubes in Group | Enveloping Tube | | | Additional Tubes in Group | | | Limiting RSR | Minimum Acceptable Steam Pressure at Feedwater Temp. (psia) | |
|---|---|---|---|---|---|---|---|---|---|---|
| | | S.G. | Row | Col. | S.G. | Row | Column | | $T_{fw}$ = 400 Deg. F | $T_{fw}$ = 455 Deg. F |
| 1 | 2 | B | 11 | 3 | B | 11 | 4 | | | a,c |
| 2 | 1 | C | 10 | 4 | none | - | - | | | |
| 3 | 1 | B | 11 | 5 | none | - | - | | | |
| 4(2) | 2 | C | 9 | 42 | C | 9 | 43 | | | |
| 5(3) | 18 | C | 10 | 88 | C | 10 | 89 through 93 | | | |
| | | | | | C | 10 | 2 through 3 | | | |
| | | | | | B | 10 | 2 through 5 | | | |
| | | | | | B | 10 | 91 through 93 | | | |
| | | | | | A | 10 | 2 through 3 | | | |
| | | | | | C | 10 | 5 | | | |
| 6(3) | 3 | B | 9 | 53 | C | 9 | 88 | | | |
| | | | | | B | 9 | 90 | | | |
| 7(3) | 7 | C | 8 | 25 | B | 8 | 34 | | | |
| | | | | | C | 8 | 26 | | | |
| | | | | | B | 8 | 60 | | | |
| | | | | | C | 8 | 60 | | | |
| | | | | | A | 8 | 35 | | | |
| | | | | | C | 8 | 18 | | | |
| 8(3) | 1 | B | 11 | 2 | none | - | - | | | |
| | | | | | | | | | | |

Note:

(1) Tubes listed should be removed from service should the normal operating steam pressure fall to the value indicated at the corresponding feedwater temperature.

(2) This Tube Group and tubes do not require action since the "minimum acceptable steam pressure" is below the minimum steam pressure of 700 psia required for full power operation at EPU conditions.

(3) This Tube Group and tubes do not require action since the "minimum acceptable steam pressure" is below that shown for the EPU cases (See Table 4.7.2.3-1).

#### 4.7.2.4 Hardware Changes and Additions

This section summarizes the structural evaluations of steam generator hardware changes and additions, which are inservice repairs. The evaluations are performed to qualify existing or future repairs for the EPU loading conditions at BVPS-2. Such inservice repairs must satisfy the rules and requirements of Article IWA-4000 in Section XI of the ASME Code, Reference 1. Subsection IWA-4120(a) of Section XI permits the criteria and material data from Section III of the ASME Code to be employed in the evaluation.

The original loading conditions are defined in the design specifications for the steam generators. The EPU results in changes to the specified normal and upset transients. These changes to the normal and upset transients have been considered in this evaluation. The faulted and test conditions are unaffected by the EPU and remain as specified in the design specifications. In some cases, the original faulted or test loads bound the structural evaluation rather than the EPU. These cases are also identified herein. There are no emergency conditions specified in the design specifications.

##### 4.7.2.4.1 Tube Mechanical Plug

Two lengths of mechanical plugs have been used, 4.262 and 2.562 inches, "long" and "short" mechanical plugs. The ribbed sealing land region geometry of both plugs is identical. The "long" mechanical plugs are fabricated from Alloy 690 rod, with a minimum specified yield strength of 35 ksi. The "short" mechanical plugs are fabricated from either Alloy 600 or 690 rod, both with minimum yields of 35 ksi. Any Alloy 600 mechanical plugs that remain in the steam generators have the "plug-in-plug" feature.

The original generic structural analysis qualified these mechanical plugs for loads that bound the conditions for BVPS-2. The generic analysis was extended for the revised load conditions and transients, which correspond to the EPU conditions. That evaluation addresses both plug lengths and both materials (Alloys 600 and 690). Since the specified minimum strengths and fatigue design curves are the same, the structural allowables, based on the ASME Code criteria, are the same for all plug materials. The generic and EPU evaluations employ acceptance criteria that include both plug retention and compliance with the ASME Code structural rules from References 2 and 3.

Plug retention is demonstrated if the strain at the tube to tubesheet interface, produced by the mechanical plug due to installation (preload), exceeds the unloading strain produced by tubesheet hole dilation resulting from tubesheet bowing under the various specified loading conditions. It is also necessary to show that there is adequate friction to prevent dislodging of the plug for the limiting transient. Two regions of the tubesheet are limiting with respect to plug retention and contact integrity. The first is near the center of the tubesheet, where the limiting load occurs for the secondary side hydrostatic test with a primary-to-secondary $\Delta P$ of -1356 psi across the tubesheet. The second region is near the periphery of the tubesheet, where the limiting load occurs for the primary side hydrostatic test with a primary-to-secondary $\Delta P$ of 3106 psi. It has been shown that both of these regions and conditions satisfy the plug retention acceptance criteria. While the hydrostatic tests are unlikely to be repeated, they bound all subsequent service (normal, upset, test) and postulated faulted conditions for plug contact integrity and retention.

In the generic ASME Code structural evaluation of the mechanical plug shell region above the uppermost sealing land, an overall maximum internal to external ΔP of 2485 psi, equal to the primary side design pressure, was assumed. Since this value exceeds all expected internal to external ΔPs for normal and upset conditions due to the EPU, the mechanical plug is structurally acceptable for the EPU pressure loads. Recall that test and faulted pressure loads, which are unaffected by the EPU, were evaluated in the original generic analysis. Also, the generic fatigue evaluation of the mechanical plug for the Model 51 steam generator applications demonstrated that the plug meets the fatigue exemption conditions of NB-3222.4(d) of the 1989 Code, Reference 3. These exemption conditions are the same as those in Article 415.1 of the 1971 Code, Reference 2. These fatigue exemption conditions were re-investigated for the EPU conditions and continue to be satisfied. Thus the mechanical plugging applications at BVPS-2 meet the ASME Code cyclic load fatigue limits for the EPU conditions.

Any existing Alloy 600 plugs, with the "plug-in-plug" (PIP) feature, have also been qualified for use in Model 51 steam generators. The PIP is installed in Alloy 600 mechanical plugs to prevent tube damage due to potential release of the plug top and also prevent significant leakage, should the plug develop a through wall crack (above the sealing lands). The Alloy 600 PIP has a threaded shaft, which is screwed into the expander's threaded inner diameter until a flanged head on the bottom of the PIP contacts the bottom of the plug at the tube mouth and a preload torque is established. The PIP is then locked in place with a fusion tack weld between the PIP's flanged head and the mechanical plug's shell. Pressure loads have a negligible effect on the PIP. Also, tubesheet bowing and tube hole deformation, which do affect the plug at the contact lands, have little effect on the PIP. Only the preload and the subsequent thermal transients have effects that were considered in the existing analysis. The preload is independent of the EPU, and the primary side thermal transients at the tube mouth (primary face of the tubesheet) have not changed significantly for the EPU. Thus, it was concluded that the existing qualification of the PIP, for installation in Model 51 steam generators, remains valid for the EPU at BVPS-2.

#### 4.7.2.4.2 Tube Rolled Plug and Tube Stabilizers

Alloy 690 rolled plugs manufactured by Framatome ANP are installed in the steam generators. The equipment specifications for the Framatome ANP rolled plugs were updated to address the BVPS-2 EPU loads. Subsequent calculations and analyses performed by Framatome ANP confirm that the rolled plugs meet all applicable design criteria and are acceptable for operation at 2910 MWt.

Stabilizers manufactured by Framatome ANP are installed at BVPS-2. Three types of stabilizer designs are utilized. One design is a sleeve type stabilizer fabricated from Inconel 690 material. A segmented stabilizer design fabricated from Inconel 600 material is also utilized as well as a cable style stabilizer fabricated from stainless steel (316L). The equipment specifications for the Framatome ANP stabilizers were updated to address the BVPS-2 EPU loads. Subsequent calculations and analyses performed by Framatome ANP confirm that all three designs of tube stabilizers meet all applicable design criteria and are acceptable for operation at 2910 MWt.

#### 4.7.2.4.3 Tube Weld Plug

Two types of weld plugs are installed, the NPT-80 and the NPT-23. Both weld plugs require that the ends of the steam generator tubes be machined to accept the weld plug. The NPT-80 weld plugs are fabricated from Alloy 690 rod, with a minimum specified yield strength of 35 ksi. The NPT-23 weld plugs are fabricated from Alloy 600 rod, also with a minimum specified yield of 35 ksi. The fatigue design curves for Alloy 690 and Alloy 600 are identical. Therefore, the structural allowables, based on the ASME Code criteria, are the same for both types of weld plugs. The NPT-80 weld joint and plug wall sections are smaller than those of the solid NPT-23 plug. Therefore, for the same loads and structural allowables, the primary stress and fatigue evaluations of the NPT-80 weld plug envelope those of the NPT-23 plug for the 7/8 inch tube application.

All primary stress limits were satisfied for the welds between the plug and the tubesheet cladding. The generic primary stress evaluation was unaffected by the EPU and remains valid for design, faulted and test conditions. The overall maximum primary plus secondary stress in the weld for normal and upset loads, adjusted for the EPU, was found to be [ ]$^{a,c}$ ksi, compared to an allowable of 31.4 ksi. The 31.4 ksi allowable conservatively included a weld quality factor of 0.45, as assumed in the generic evaluation.

The fatigue usage factor, recalculated for the EPU, remained low at [ ]$^{a,c}$, well below the ASME Code allowable of one. The cumulative usage factor was conservatively recalculated for the EPU using a fatigue strength reduction factor (FSRF) of four, as assumed in the generic usage calculations. These results, which are based on an explicit calculation of the stress ranges due to the EPU and a conservative FSRF of 4, confirmed the results from the mechanical plug fatigue evaluation based on satisfying the ASME Code fatigue exemption conditions, as applied at the primary face of the tubesheet. Recall that the mechanical plugs and weld plugs are fabricated from the same materials, are located in the same region, and experience the same thermal transients. Therefore, a low explicitly calculated fatigue usage factor for the weld plug would be expected even using a conservatively large FSRF.

#### 4.7.2.4.4 Sleeved Tube Mechanical Plug

Mechanical plugs are used to remove tubes from service. Occasionally it is necessary to plug a tube which has previously had a sleeve installed (sleeved tubes). This section summarizes the qualification of Alloy 690 mechanical plugs for installation in 7/8 inch sleeved tubes in the Model 51 steam generators at BVPS-2 for the EPU conditions. As in the tube mechanical plug, the sleeved plug is fabricated from 35 ksi minimum yield strength Alloy 690 rod. The design of the sleeved tube plug is similar to the "short" tube plug with a length of [ ]$^{a,c}$ inches (compared to [ ]$^{a,c}$ inches) and a diameter of 0.[ ]$^{a,c}$ inch (compared to [ ]$^{a,c}$ inch). As with the tube plug, the sleeved tube mechanical plug also employs ten ribs for sealing. The major difference, in the application of the sleeved tube mechanical plug (compared to the tube mechanical plug), is that the plug preload on the sealing lands must be established through the Alloy 690 sleeve ([ ]$^{a,c}$ inch nominal wall) in addition to the Alloy 600 tube (0.050 inch nominal wall). Both plug retention and compliance of the plug shell above the uppermost sealing land with the ASME Code structural criteria were considered in the qualification.

As with the tube mechanical plug, the sleeved tube plug retention performance was evaluated considering the effect of tubesheet bowing for several bounding specified loading conditions. The interaction between the tubesheet and plugs during any loading condition is affected by local size changes (dilations or contractions) of the tube holes in the tubesheet. The effects of such hole-size changes on the plug radial contact stresses were determined using global and local axisymmetric models. Table 4.7.2.4-1 shows the maximum load conditions considered in the sleeved tube plug retention evaluation for the BVPS-2 application.

Considering the specified EPU normal operation, upset transients, faulted and test conditions in Table 4.7.2.4-1, the overall limiting plug retention condition occurred for the primary side hydrostatic pressure test in which the ΔP across the plug was found to be 3106 psi. Due to the calculated tubesheet bowing deformations (including the effects of the channel head and main cylinder), plugs installed in the outer most tubes would experience the maximum reduction in the initial preload (the radial contact pressure at the plug lands due to installation). For this limiting load at the most limiting location in the tubesheet, and for a conservatively estimated [ ]a,c ksi initial preload contact pressure at the lands, the calculated maximum coefficient of friction required to retain the plug was found to be [ ]a,c. The actual "friction coefficient" available at the plug lands is expected to be much higher (on the order of one or more) since the plug lands plastically press into the tubes at installation. Therefore, substantial sleeved tube plug retention margin is available, even for the most limiting retention conditions.

In addition to plug retention, the sleeved tube plug shell was also qualified structurally for pressure stress and fatigue by direct comparison to the mechanical plug shell evaluated for the EPU (see Section 4.7.2.4.1). The mechanical plug for 7/8 inch tubes bounds the mechanical sleeved tube plug for 7/8 inch tubes with respect to pressure stress evaluation and fatigue. The governing pressure stress intensity in a cylindrical shell is proportional to the R/t ratio, where R is the inner radius and t is the wall thickness of the shell. Since the sleeve plug has a smaller radius (with about the same wall thickness), the pressure stress intensity in the sleeved plug wall is about 90% of the stress intensity in the tube plug for the same pressure load. In Section 4.7.2.4.1, the tube plug was also shown to meet the ASME Code fatigue exemption conditions for the EPU. Since the exemption conditions are dependent only on loading conditions and material properties (both of which are identical for the sleeve plug), then the sleeve plug also satisfies the ASME Code exemption conditions. Therefore, it is concluded that all ASME Code pressure and fatigue structural limits are also satisfied with positive margins in the sleeved tube application for the EPU at BVPS-2.

#### 4.7.2.4.5 Straight Leg Collar-Cable Stabilizer

The straight leg stabilizer is inserted in a degraded host tube to prevent the host tube from contacting the adjacent active tubes. The Westinghouse stabilizer design consists of a central coaxial stainless steel cable protected over the full length of the stabilizer by several 304 SS tubular collars, which are swaged on to the cable. The swaged collars are about [ ]a,c inches in length and are spaced at about [ ]a,c inch intervals (i.e., there is a longitudinal space of about [ ]a,c inch between adjacent collar segments to provide flexibility and dynamic damping). Two stabilizer sizes, 0.687 inch OD and 0.625 inch OD, are qualified for use in the 7/8 inch OD (0.775 inch ID) Alloy 600 tubes in BVPS-2 steam generators. The qualification method employed herein is essentially independent of loading conditions and depends only on cross-sectional areas and the contacting materials. Therefore, all existing (installed) or future (to be installed) 0.687 inch stabilizers are also qualified for the EPU conditions at BVPS-2.

The qualification method shows that the wall of an assumed fully severed host tube, would wear out before the stabilizer collar, should a random wear couple form between the host tube and the stabilizer collar. Because the host tube wall wears away first, the central coaxial cable of the Westinghouse stabilizer remains protected by the collar remnant for the life of the installation. This conclusion is based on the relative wear coefficients and the cross-sectional areas of the tube and stabilizer and is independent of the dynamic fluid forces causing potential random vibration of the assumed severed tube. Thus, changes in fluid conditions, due to the EPU, have no effect on the ultimate protective function of the stabilizer to prevent contact with adjacent tubes.

The hypothetical worst case occurs when the degraded host tube is assumed to be fully severed and becomes dynamically unstable due to fluid flow excitation. The precise flow conditions and dynamic instability did not actually require definition except to assume they were sufficient to cause the severed host tube to begin a continuous wear couple with the stabilizer. Under these extreme conditions, the stabilizer must prevent the severed host tube from contacting the adjacent tubes. Assuming that the collar and tube wear equally (i.e., assuming the wear coefficients for the couple are equal), the tube wall wears out before the swaged 0.687 inch OD larger collar, which protects the central coaxial cable for the life of the installation and prevents contact with the adjacent tubes. For the smaller 0.625 inch OD collar case, about [ ]$^{a,c}$ % of the cable is worn ([ ]$^{a,c}$ % of the cable remains) when the tube wears out totally assuming equal wear between the tube and stabilizer. Again, contact with the adjacent tubes will not occur for this case, even with an [ ]$^{a,c}$ % worn cable. However, wear test data suggest that 304 SS collars would wear at a lower rate than the Alloy 600 tubes. Therefore, the above assumption of equal wear is conservative and both size collars (0.625 inch and 0.687 inch) are shown to be adequate to protect the central cable and prevent contact with adjacent tubes for the life of the stabilizer installation.

#### 4.7.2.4.6 Laser Welded Sleeves

An analysis has been performed to evaluate the applicability of the generic laser welded sleeving analysis for 7/8 inch diameter tube steam generators to the BVPS-2 steam generators. The purpose of the analysis was to compare the transient and operating parameters corresponding to EPU conditions to those used in the generic analysis, with the intent of confirming that the generic analysis provides a bounding analysis for BVPS-2. For those cases where the generic loads were found to not be bounding, revised calculations were performed based on the parameters corresponding to BVPS-2. In establishing the structural adequacy of the laser welded sleeves (LWS), criteria were evaluated for primary stress limits, maximum range of stress intensity and fatigue, and minimum wall thickness requirements. The revised structural analysis for LWS is documented in Reference 4.

It should be noted that the approach to demonstrate Code compliance of the 0.015 inch laser weld was to verify the structural integrity of the weld based on ASME Code Section III design-by-test requirements. These criteria, however, are biased toward the attachment of butt welded fittings, and there is no geometry specified in the Code that correlates directly to the geometry of the LWS weld joint, i.e., a weld joint that is effectively loaded in pure shear. This means that the demonstration of compliance with the Code requires interpretation of the intent of the Code authors and may be construed to be subjective. In order to resolve this circumstance, structural analyses were performed to characterize the average weld width that would be necessary to demonstrate compliance with the Code design-by-analysis requirements and to achieve estimated strengths greater than the burst resistance of the sleeve. The results from the analysis work demonstrate that an average weld width of [ ]$^{a,c}$ mils meets all of the design-by-analysis requirements (no required structural tests) of the Code for all currently available LWS sleeve and tube combinations. Based on these findings, Westinghouse has revised the field inspection procedure to additionally verify that the average width of new LWS installed sleeves is ≥ [ ]$^{a,c}$ mils. Thus, the calculations to assess structural integrity of the laser weld are based on a minimum [ ]$^{a,c}$ inch weld width.

**Primary Stress Evaluation**

The LWS structural analysis evaluated the sleeves for the limiting design, faulted, and test conditions. A comparison of the applicable pressure loads for the generic case versus the values for BVPS-2 showed the generic analysis to be bounding for each load category. Thus, the results for the pressure stress evaluations remain valid and are applicable to BVPS-2. A summary of the maximum stresses in the tube and sleeve are summarized in Table 4.7.2.4-2. Results for a [ ]$^{a,c}$ inch laser weld are summarized in Table 4.7.2.4-3.

**Maximum Range of Stress and Fatigue**

Normal, upset, and test loads were evaluated relative to the maximum range of primary plus secondary stress and fatigue. The comparison of transient cycles for each of the loading conditions for the generic analysis versus the transient cycles applicable to BVPS-2 showed that, in general, the number of applied cycles for the generic analysis to be equal to or greater than the number of applied cycles for BVPS-2 for each transient condition. Comparison of the applicable loads (temperatures and pressure gradients) found that the loads for the generic case bound the High $T_{avg}$ conditions. However, for the Low $T_{avg}$ conditions, the corresponding applied loads are slightly higher than for the generic case. Thus, a revised set of fatigue calculations was performed for the Low $T_{avg}$ set of conditions.

A summary of the maximum range of stress and fatigue usage for the tube and sleeve is provided in Table 4.7.2.4-4. The results of calculations to determine revised maximum range of stress and fatigue for the weld are also summarized in Table 4.7.2.4-4. These results show the applicable ASME Code limits to be satisfied.

**Minimum Wall Thickness (Tube Repair Limits)**

The minimum acceptable wall thickness and other recommended practices in US NRC Draft Regulatory Guide 1.121, Reference 5, were used to determine a repair limit for the sleeve. The Regulatory Guide was written to provide guidance for the determination of a repair limit for steam generator tubes undergoing localized tube wall loss and can be conservatively applied to sleeves. Tubes with sleeves that are determined to have indications of degradation of the sleeve in excess of the repair limit would have to be repaired or removed from service. As recommended in paragraph C.2.b of the Draft Regulatory Guide, an additional thickness degradation allowance must be added to the minimum acceptable sleeve wall thickness to account for eddy current uncertainty and continued degradation in order to establish the operational sleeve thickness acceptable for continued service. A summary of the required minimum wall thickness and corresponding structural limits is provided in Table 4.7.2.4-5.

#### 4.7.2.4.7 TIG-Welded Sleeves

The structural adequacy of the steam generator TIG-welded sleeve-tube assembly was evaluated for operation at EPU conditions including lower feedwater temperature. Lower feedwater temperature could possibly create a higher temperature difference across the tube wall than previously evaluated in the current generic licensing reports for the TIG-welded sleeves (References 6 and 7). The current generic licensing reports evaluated axial loads in the sleeve for the most severe combinations and the "worst" case conditions for Westinghouse plants with 7/8 inch Inconel 600 tubes. In References 6 and 7, the maximum axial load of [ ]$^{a,c}$ lbs. was calculated for 100% steady state power with the tube not locked into the tube support. The minimum test value was [ ]$^{a,c}$ lbs. for the rolled section. This provides a minimum safety factor of 1.41.

The maximum axial loads on the TIG-welded sleeve-tube assembly in the BVPS-2 steam generators operating at EPU full power conditions, feedwater temperature as low as 400°F, and RCS hot leg temperature as high as 617.0°F is [ ]$^{a,c}$ lbs. This value is lower than [ ]$^{a,c}$ lbs. reported for the "worst" case Westinghouse plant in the TIG-welded sleeve generic licensing reports. Therefore, the axial loads associated with operation of the BVPS-2 steam generators at EPU conditions with a feedwater temperature as low as 400°F remain bounded for all tube locations by the generic licensing reports (References 6 and 7) for TIG-welded sleeves.

At EPU conditions with the minimum allowable secondary pressure of [ ]$^{a,c}$ psia (BVPS-2 Case 1a in Section 4.7.2.1), the new sleeve wall allowable degradation was calculated to be [ ]$^{a,c}$ % vs. the [ ]$^{a,c}$ % in the TIG-welded sleeve generic licensing reports (References 6 and 7). As recommended in paragraph C.2.b of U.S. NRC Draft Regulatory Guide 1.121 (reference 5), an additional thickness degradation allowance must be added to the minimum acceptable sleeve wall thickness to account for eddy current uncertainty and continued degradation in order to establish the operational sleeve thickness acceptable for continued service.

Based on the analysis for TIG-welded sleeves at EPU conditions, the requirements stipulated in the generic licensing reports for TIG-welded sleeves (References 6 and 7) remain satisfied for their use in the BVPS-2 steam generators at EPU conditions, with the exception of the sleeve wall allowable degradation value which is revised to [ ]$^{a,c}$ % not including the additional thickness degradation allowance to account for eddy current uncertainty and continued degradation.

#### 4.7.2.4.8 Conclusions

The results and conclusions of the analyses and evaluations performed for steam generator hardware changes and additions for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

#### 4.7.2.4.9 References


1. ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY, (Latest edition applicable for use at BVPS-2).

2. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY, 1971 Edition plus Addenda through Summer 1972 (Code of Construction for BVPS-2).

3. ASME Boiler and Pressure Vessel Code, Section III, "Rules for Construction of Nuclear Power Plant Components," American Society of Mechanical Engineers, New York, NY, 1989 Edition.

4. WCAP-13483, Revision 2, "Beaver Valley Units 1 and 2 Westinghouse Series 51 Steam Generator Sleeving Report, Laser Welded Sleeves," October 2002.

5. US NRC Draft Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes (For Comment)," August 1976.

6. CEN-629-P, Rev. 2, "Repair of Westinghouse SERIES 44 & 51 S/G Tubes Using Leak Tight Sleeves," January 1997.

7. CEN-629-P, Addendum 1, "Repair of Westinghouse SERIES 44 & 51 S/G Tubes Using Leak Tight Sleeves," January 1997.

**Table 4.7.2.4-1**
**BVPS-2 Maximum Load Conditions Considered in Sleeved Tube Plug Retention Evaluation**
**Conditions 3 and 4 (Normal and Upset) Incorporate the EPU**
**(All Other Load Conditions are Unaffected by the EPU)**

| # | Load Conditions | Pp (psig) | Tp (°F) | Ps (psig) | Ts (°F) | ΔP = Pp – Ps (psi) |
|---|---|---|---|---|---|---|
| # | **DESIGN** | | | | | |
| 1 | Primary Side | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| 2 | Secondary Side | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| | **MAXIMUM NORMAL ΔP** | | | | | |
| 3 | 10% Small Step Increase (79.6 s)(1) | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ (1) |
| | **MAXIMUM UPSET ΔP** | | | | | |
| 4 | Loss of Flow (13.8 s) | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| | **FAULTED** | | | | | |
| 5 | Reactor Coolant Pipe Break | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| 6 | Feed Line Break | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| | **TEST** | | | | | |
| 7 | Primary Hydrostatic Test | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ |
| 8 | Secondary Hydrostatic Test | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ |
| 9 | Tube Leak Test | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ |
| 10 | Primary Side Leak Test | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ | [ ]$^{a,c}$ (2) | [ ]$^{a,c}$ |
| 11 | Secondary Side Leak Test | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |

Notes:

(1) The overall maximum normal primary-to-secondary pressure differential absolute value of [ ]$^{a,c}$ psi occurs for the 10% small step load increase transient at time 79.6 seconds for low $T_{avg}$ and 22% steam generator tube plugging in the EPU. However, the minimum acceptable full power steam pressure is limited to 685 psig (700 psia) to keep the maximum normal ΔP from exceeding the 1600 psi design limit (load # 1). Thus, the [ ]$^{a,c}$ psi differential is not an actual load condition. However, the sleeve plug evaluation performed for the EPU considered the [ ]$^{a,c}$ psi differential load herein to be conservative.

(2) Since higher temperatures tend to increase the plug/sleeve contact pressure and increase the plug retention forces, these tests were assumed to occur at an ambient temperature ([ ]$^{a,c}$) giving a lower (conservative) estimate of the plug/sleeve contact pressures.

| Table 4.7.2.4-2<br>BVPS-2 Summary of Maximum Primary Stress Intensity<br>Full Length Tubesheet Laser Welded Sleeve<br>Tube Intact | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| **Design** | | | | | | | | | |
| **Location** | **$P_m$** | **Allowable** | **Ratio** | **$P_l + P_b$** | **Allowable** | **Ratio** | **$S_1+S_2+S_3$** | **Allowable** | **Ratio** |
| Sleeve | [ ]$^{a,c}$ | 26.60 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 39.90 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 106.40 | [ ]$^{a,c}$ |
| Tube | [ ]$^{a,c}$ | 23.30 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 34.95 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 93.20 | [ ]$^{a,c}$ |
| **Faulted** | | | | | | | | | |
| Sleeve | [ ]$^{a,c}$ | 56.00 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 84.00 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 106.40 | [ ]$^{a,c}$ |
| Tube | [ ]$^{a,c}$ | 55.92 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 83.88 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 93.20 | [ ]$^{a,c}$ |
| **Testing** | | | | | | | | | |
| Sleeve | [ ]$^{a,c}$ | 36.00 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 54.00 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 106.40 | [ ]$^{a,c}$ |
| Tube | [ ]$^{a,c}$ | 31.50 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 47.25 | [ ]$^{a,c}$ | [ ]$^{a,c}$ | 93.20 | [ ]$^{a,c}$ |

| Table 4.7.2.4-3<br>BVPS-2 Summary of Maximum Primary Stress Intensity<br>0.021 inch Average Laser Weld | | | |
|---|---|---|---|
| **Loading** | **Calculated Stress (ksi)** | **Allowable Stress (ksi)** | **Ratio of Calculated to Allowable** |
| Design | [ ]$^{a,c}$ | 15.96 | [ ]$^{a,c}$ |
| Upset | [ ]$^{a,c}$ | 17.56 | [ ]$^{a,c}$ |
| Test | [ ]$^{a,c}$ | 16.79 | [ ]$^{a,c}$ |
| Faulted | [ ]$^{a,c}$ | 33.6 | [ ]$^{a,c}$ |

| Table 4.7.2.4-4<br>BVPS-2 Summary of Maximum Range of Stress Intensity and Fatigue<br>Tube and Sleeve Tube Severed and Dented | | | |
|---|---|---|---|
| **Component** | **Calculated Stress Intensity (ksi)** | **Allowable Stress Intensity (ksi)** | **Margin to Allowable** |
| Sleeve | [ ]$^{a,c}$ | 79.80 | [ ]$^{a,c}$ |
| Tube | [ ]$^{a,c}$ | 69.90 | [ ]$^{a,c}$ |
| Weld | [ ]$^{a,c}$ | 79.80 | [ ]$^{a,c}$ |
| **Cumulative Fatigue Usage Factor** | | | |
| Tube and Sleeve | | [ ]$^{a,c}$ << 1.0 | |
| Weld | | [ ]$^{a,c}$ < 1.0 | |

| Table 4.7.2.4-5<br>BVPS-2 Summary of Sleeve Minimum Wall Thickness Requirements and<br>Corresponding Tube Structural Limits | | | |
|---|---|---|---|
| **Quantity** | **Generic** | **High $T_{avg}$** | **Low $T_{avg}$** |
| Required Wall Thickness (inch) | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Structural Limit (%)$^{(1)}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Note:<br>(1) Structural Limit = $[(t_{nom} - t_{min}) / t_{nom}] \times 100\%$ | | | |

### 4.7.2.5 Tube Wear

#### 4.7.2.5.1 Introduction

The impact of the EPU on steam generator tube wear was evaluated based on the known wear history for the last several outages. This wear data was provided in the form of eddy current test data percent through-wall indications taken during successive outages, and is converted into wall penetration depths. These depths were then converted in wear volumes. The difference between these volumes over time is the volumetric wear rate experienced between the times when the readings were taken. This volumetric wear rate is then converted into a volumetric wear rate per effective full power year. The effect of EPU on the wear rate is to potentially accelerate the wear rate due to higher flow energy. Any change to the thermal-hydraulic parameters may result in a change in the wear rate. To account for these effects, an EPU wear factor was calculated which was used to modify the wear rates calculated, thereby accounting for the potential acceleration in the wear rate that resulted from the EPU.

Wear in the steam generator tubes due to flow effects is more likely to occur in the U-bend region of the tubes. An evaluation was performed to extrapolate the wear on the U-bend tubes to assess the consequences for plugging, and tube-to-tube contact, both with and without the EPU. This extrapolation was based on the volumetric wear rate calculated based on historic eddy current data, both with and without modification, to account for the EPU; and the future predicted availability of the plant, to estimate when wear depth limits are reached.

#### 4.7.2.5.2 Description of Analyses and Evaluations

The tube population in the steam generators that show anti-vibration bar (AVB) wear, both current and plugged, is approximately 60 tubes for BVPS-2. This is from a total tube population for the three steam generators of 10,128 tubes. This represents approximately 0.6% of the tubes. A review of the 2R10 eddy current test (ECT) data for BVPS-2 shows that 55 of the tubes exhibit AVB wear.

Without an EPU, as many as 23 additional tubes might have to be plugged for BVPS-2 prior to the end of the steam generator service lives. This number increases to 25 tubes plugged for BVPS-2 with an EPU, an increase of 2 tubes plugged for BVPS-2. At the 2R10 outage, no tubes exhibited AVB wear depth that exceeded the Technical Specification repair limit of 40%.

There were no tubes that are projected to initiate tube-to-tube contact, with or without the EPU based on a 30% through-diameter wear limit.

#### 4.7.2.5.3 Results and Conclusions

The ECT results show that wear due to flow induced vibration for BVPS-2 is low as evidenced by the small number of tubes that exhibit AVB wear. Even though a notable increase ([ ]$^{a,c}$ % for BVPS-2) in wear is predicted for the EPU, the net effect on potential currently active tube plugging is small. It can also be concluded that since the thermal-hydraulic effects that result in tube vibration and subsequent wear are small in the tube U-bend region, it would also be small in the straight leg region of the tubes. Evidence to support this conclusion can be seen in current, more detailed, analysis of feeding steam generators. Results show that the greatest amount of predicted tube wear occurs in the U-bend region of the tubes. This is further supported by the higher calculated instability ratio in the U-bend region of the tubes. Therefore, because it is shown that the effect of the EPU does not significantly affect the U-bend region, it can also be concluded that the effect along the straight tube region will also not be significant.

Tubes that currently do not show any reportable tube wear demonstrate at most a very low rate of wear. For Cycle 10, the average AVB wear growth rate was [ ]$^{a,c}$% per EFPY. Increasing this low rate of wear by as much as [ ]$^{a,c}$% will not result in a significant increase in tube plugging since the increase is of a small wear rate.

It can, therefore, be concluded that the EPU will not have a significant impact on tube wear in the BVPS-2 steam generators. The increase in tube wear will not result in a significant increase in tube plugging levels due to tube vibration and so will not adversely affect the operation of the steam generators.

The results and conclusions of the analyses and evaluations performed for steam generator tube wear for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

### 4.7.2.6 Tube Plugging or Repair Limit (Draft Regulatory Guide 1.121 Analysis)

#### 4.7.2.6.1 Introduction

The heat transfer area of steam generators in a PWR nuclear steam supply system (NSSS) comprises over 50% of the total primary system pressure boundary. The steam generator tubing, therefore, represents a primary barrier against the release of radioactivity to the environment. For this reason, conservative design criteria have been established for the maintenance of tube structural integrity under the postulated design-basis accident condition loadings in accordance with Section III of the ASME Code.

Over a period of time under the influence of the operating loads and environment in the steam generator, some tubes may become degraded in local areas. Partially degraded tubes are satisfactory for continued service provided that defined stress and leakage limits are satisfied, and that the prescribed structural limit is adjusted to take into account possible uncertainties in the eddy current inspection, and an operational allowance for continued tube degradation until the next scheduled inspection.

#### 4.7.2.6.2 Description of Analyses and Evaluations

The US NRC Draft Regulatory Guide 1.121, Reference 1, describes an acceptable method for establishing the limiting safe condition of degradation in the tubes beyond which tubes found defective by the established in-service inspection shall be removed from service. The level of acceptable degradation is referred to as the "plugging or repair limit."

An analysis has been performed to define the "structural limits" for an assumed uniform thinning mode of degradation in both the axial and circumferential directions using ASME Code minimum material properties. The assumption of uniform thinning is generally regarded to result in a conservative structural limit for all flaw types occurring in the field. The allowable tube plugging or repair limit, in accordance with Draft Regulatory Guide 1.121, is obtained by incorporating into the resulting structural limit a growth allowance for continued operation until the next scheduled inspection and also an allowance for eddy current measurement uncertainty. Calculations have been performed to establish the structural limit for the tube straight leg (free-span) region of the tube for degradation over an unlimited axial extent, and for degradation over limited axial extent at the tube support plate and AVB intersections.

The analysis includes a specification on maximum allowable leak rate during normal operation consistent with the EPRI PWR Primary-to-Secondary Leak Guidelines (Reference 2). These guidelines define several monitoring and action level conditions, depending on the amount of leakage and the rate of leakage increase.

The analysis includes steamline break (SLB) loads for a SLB peak transient differential pressure of 2485 psi consistent with the steam generator design specification. This is conservative since BVPS-2 operates with the pressurizer power operated relief valves available which limits the SLB peak transient differential pressure to 2405 psi. The SLB loads based on a peak transient differential pressure of 2485 psi are conservative for a peak transient differential pressure of 2405 psi.

#### 4.7.2.6.3 Acceptance Criteria and Results

A summary of the resulting tube structural limits is provided in Table 4.7.2.6-1.

#### 4.7.2.6.4 Conclusions

A Draft Regulatory Guide 1.121 analysis was performed to establish the structural and tube plugging or repair limits for the steam generator tubing at BVPS-2. The results of the analysis allow the utility to establish a steam generator maintenance program that will preserve the structural integrity of the steam generators.

The results and conclusions of the analyses and evaluations performed for the steam generator tube plugging or repair limit for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

#### 4.7.2.6.5 References

1. US NRC Draft Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes (For Comment)," August 1976.

2. EPRI Report TR-104788-R2, "PWR Primary-To-Secondary Leak Guidelines – Revision 2," EPRI, Palo Alto, CA, 2000.

| Table 4.7.2.6-1<br>Summary of Tube Structural Limits | | | | | |
|---|---|---|---|---|---|
| Location/<br>Wear Scar Length | Parameter | High $T_{avg}$ | | Low $T_{avg}$ | |
| | | 0% Plugging | 22% Plugging | 0% Plugging | 22% Plugging |
| Straight Leg (>1.5") | $t_{min}$ (inch) | | | | a,c |
| | Structural Limit (%)* | | | | |
| AVB/0.60"** | $t_{min}$ (inch) | | | | |
| | Structural Limit (%)* | | | | |
| TSP/0.75" | $t_{min}$ (inch) | | | | |
| | Structural Limit (%)* | | | | |

AVB = Antivibration Bar

TSP = Tube Support Plate

* Structural Limit = $[(t_{nom} - t_{min})/t_{nom}] \times 100\%$, $t_{nom}$ = [ ]$^{a,c}$ inches

** Tube structural limits are provided for an AVB wear scar length of 0.60". The actual AVB wear scar length is expected to be bounded by a wear scar length of 0.45". The tube structural limits for a 0.60" wear scar are conservative for a 0.45" wear scar.

#### 4.7.2.7 Tube Degradation

##### 4.7.2.7.1 Introduction

Potential tube degradation mechanisms including primary water stress corrosion cracking (PWSCC) and outside diameter stress corrosion cracking (ODSCC) were evaluated for BVPS-2 for operation at the operating temperatures (i.e., best estimate $T_{hot} \leq 611°F$) associated with EPU conditions.

Since there are no current plans for steam generator replacement for BVPS-2, tube degradation mechanisms were considered to the end of the original license. The degradation mechanisms addressed for EPU include the BVPS-2 2R10 inspection results.

The BVPS-2 steam generators use mill annealed Alloy 600 tubing, drilled hole carbon steel support plates, and full depth mechanically (roll) expanded tubes in the tubesheet region. The BVPS-2 steam generator tube expansion transitions at the top of tubesheet were shotpeened prior to operation, and the small radius (Row 1 and 2) U-bends were heat treated prior to operation. Additionally, the use of a flow distribution baffle in the BVPS-2 design has limited sludge deposition on the top of tubesheet. The BVPS-2 steam generators contain 3376 original tubes. The tube count for BVPS-2 is reduced from that in similar generators (e.g., BVPS-1 Model 51 OSGs, 3388 original tubes) due to the inclusion of several additional stayrods. These stayrods effectively reduce potential tube support plate displacement during postulated accident conditions.

All crack-like indications in the sludge pile region, expansion transition region, tubesheet region, and small radius U-bends are repaired upon detection. AVB wear and cold leg thinning indications are repaired per the current Technical Specification tube plugging criterion, although cold leg thinning has not been observed at BVPS-2. Alternate repair criteria per GL 95-05 for tube support plate (TSP) intersections are licensed for BVPS-2; however, application of the criteria has not been required due to the extremely low number of potential indications reported at BVPS-2 and low rotating probe coil (RPC) confirmation rate.

##### 4.7.2.7.2 Description of Tube Degradation Mechanisms

The most recent steam generator inspection of BVPS-2 (2R10 outage) was performed during the Fall of 2003. This inspection program included 100% inspection of the hot leg top of tubesheet region with +Pt, 100% full-length bobbin inspection, and 100% inspection of the Row 1 and 2 U-bends with the mid-range +Pt coils. Due to detection of circumferential PWSCC indications at Row 3 to Row 10 U-bends at another plant, 100% of the Row 3 to Row 10 and 20% of the Row 12 to Row 18 U-bends were also inspected with +Pt at 2R10. No crack-like indications were detected. At the 2R08 inspection (Fall 2000), Row 1 and 2 U-bends were inspected using both the mid-range and high frequency +Pt coils. No crack-like indications were detected.

For the accumulated service period of BVPS-2 (12.6 EFPY), tube plugging is low compared to other operating steam generators with similar tube material and assembly practices. The current plugging levels in the BVPS-2 steam generators are 4.06% in steam generator A, 3.73% in steam generator B, and 3.61% in steam generator C, for an overall plugging level of 3.8%. Only about 1/3 of the tubes plugged at BVPS-2 are plugged due to SCC mechanisms.

#### 4.7.2.7.2.1 Top of Tubesheet ODSCC Degradation Mechanisms

The BVPS-2 tubes are full depth mechanical (roll) expanded throughout the tubesheet thickness. This process results in differing residual stresses at the tubesheet expansion transition region compared to the BVPS-1 OSGs. As a result, the dominant mechanism for tube plugging at BVPS-2 is circumferentially oriented ODSCC within the tube expansion transition at the top of the tubesheet. This trend is consistent with the other operating steam generators that use this tube expansion technique.

The number of tubes reported with circumferential ODSCC at the expansion transition at the 2R10 inspection was slightly increased compared to previous outages (23 tubes at 2R10 versus 20 tubes at 2R09). In situ pressure testing of top of tubesheet indications was not performed at the 2R10 outage. Flaw +Pt amplitudes, arc lengths, and maximum depths were bounded by previous inspection results and in situ pressure testing. In situ pressure testing was performed for the limiting subset of circumferential ODSCC indications at the 2R06 and 2R08 outages, even though no indications were required to be pressure tested based on the reported flaw parameters. No primary-to-secondary leakage or burst was reported during in situ pressure testing.

Circumferential ODSCC growth rates for Cycle 10 were found to be consistent with previous cycles. Circumferential ODSCC structural integrity is dependent upon the observed percent degraded area (PDA), or flawed area compared to the full tube cross sectional area. For Cycle 10, based on a 548 EFPD operating period, the largest PDA growth was found to be [ ]$^{a,c}$ %, while the largest reported PDA for any flaw at 2R10 was [ ]$^{a,c}$ %. The structural limit associated with circumferential ODSCC is approximately [ ]$^{a,c}$ % PDA.

PDA growth following EPU is expected to remain less than 16% based on application of an Arrhenius equation using the current and projected operating temperature post-EPU.

As with the BVPS-1 OSGs, all BVPS-2 ODSCC indications have been located within the bounds of the historic sludge pile region, where the sludge pile region is defined based on detection of sludge-like signals using low frequency bobbin analysis. The BVPS-2 steam generator design includes a flow distribution baffle, which is designed to provide a sweeping flow across the tubesheet, thus reducing the sludge pile extent. In BVPS-2, approximately 300 tubes per steam generator are located within the sludge pile region, whereas approximately 700 tubes per steam generator were located within the sludge pile of the BVPS-1 OSGs (prior to chemical cleaning).

As with the circumferential ODSCC mechanism, the axial ODSCC at top of tubesheet degradation at BVPS-2 has been located within the historic sludge pile. Axial ODSCC flaw parameters for BVPS-2 are bounded by the BVPS-1 OSG axial ODSCC degradation, in length, +Pt amplitude, and observed growth values.

Should a significant increase in observed ODSCC growth rates be detected following EPU, the increased growth rates will be considered in the condition monitoring/operational assessment.

Long term degradation projection of the BVPS-2 top of tubesheet ODSCC mechanisms was performed based on the reported ODSCC history to date, adjusted for temperature. Using a Log-Normal curve fit, the plugging contribution due to ODSCC mechanisms at the top of tubesheet is projected to be approximately 14% at end of current license. The Log-Normal curve fit is slightly conservative compared to the Weibull Minima fit. This is expected to be a conservative approximation since the ODSCC detected to date has been confined to the historic sludge pile region. If all tubes in this (sludge pile) region eventually develop ODSCC, the plugging impact is approximately 8%. The BVPS-1 OSG post-chemical cleaning inspection data for the 1R14 inspection (15.7 accumulated EFPY) showed that all reported ODSCC remained within the bounds of the historic sludge pile, and no ODSCC was reported outside of this region after removal of the sludge pile deposits.

#### 4.7.2.7.2.2 Tubesheet Region PWSCC Degradation Mechanisms

The BVPS-2 tubesheet and expansion transitions regions of the tubes were shotpeened prior to operation. As a result, no PWSCC in the tubesheet region was reported at BVPS-2 through the Cycle 8 inspection. At the 2R09 outage, 1 tube was reported with a short, shallow axial PWSCC indication at the expansion transition. No PWSCC in the tubesheet region was reported at the 2R10 outage.

Similar plants that have performed peening prior to operation have reported consistent inspection results. An operating plant with Model D4 steam generators (mill annealed Alloy 600 tubing, hardrolled tube expansion) that operates at approximately 621°F and also shotpeened the expansion transitions prior to operation, reported a single PWSCC indication at the Spring 2001 inspection (8.99 accumulated EFPY). This was the first reported PWSCC at this plant. At the Fall 2002 inspection, two tubes were reported with PWSCC at the expansion transition and at the Spring 2004 inspection, 4 tubes were reported with PWSCC at the expansion transition (10.3 accumulated EFPY). The operating temperature of this plant with Model D4 steam generators represents a 47% higher PWSCC initiation potential compared to BVPS-2 at the operating temperatures for EPU conditions. Therefore, PWSCC mechanisms are not expected to be significantly affected by operation at EPU conditions.

Circumferential PWSCC has not been a significant plugging contributor in hardroll expanded plants. The application of shotpeening prior to operation should further help to reduce circumferential PWSCC initiation in BVPS-2. Circumferential PWSCC at the expansion transition or expanded tube length below the expansion transition region has not been reported to date at BVPS-2.

#### 4.7.2.7.2.3 Small Radius U-bend PWSCC Degradation Mechanisms

The BVPS-2 Row 1 and 2 small radius U-bend regions were heat treated prior to operation. As a result, no PWSCC at small radius U-bends has been reported at BVPS-2 through the Cycle 10 inspection. Similar plants that have performed heat treatment prior to operation have reported consistent inspection results. An operating plant with Model D4 steam generators (mill annealed Alloy 600 tubing, Row 1 bend radius approximately equal to BVPS-2) that operates at approximately 621°F and also heat treated the small radius U-bends prior to operation has not reported PWSCC at this location through 10.13 accumulated EFPY. Another plant with Model D4 steam generators that heat treated the small radius U-bends prior to operation has not reported PWSCC at this location through 10.8 accumulated EFPY. This plant initially operated at a $T_{hot}$ value of 618°F, but reduced $T_{hot}$ to 612°F several cycles ago.

Based on the operating temperatures of these other plants and current operating history of BVPS-2, small radius U-bend PWSCC is not expected to be observed before the 2R12 (Fall 2006) inspection.

#### 4.7.2.7.2.4 Tube Support Plate ODSCC

Tube support plate ODSCC at BVPS-2 has not been a significant mechanism. At the 2R10 inspection, only 491 bobbin indications were reported, of which only 3 were confirmed as axial ODSCC by +Pt examination. The number of bobbin indications was slightly greater than the value reported at 2R09 (329) and 2R08 (279). Maximum bobbin amplitude reported at 2R10 was [ ]$^{a,c}$ volts. Average bobbin voltage growth for Cycle 10 was approximately [ ]$^{a,c}$ volts ([ ]$^{a,c}$ volts per EFPY). Average bobbin voltage growth for Cycle 9 was slightly negative ([ ]$^{a,c}$ volts), and can be assumed to be 0. As the voltage based repair criteria per GL 95-05 is licensed for BVPS-2 but has not been implemented, should initiation or growth be affected by EPU such that significant increases in bobbin indication count or confirmed indications are noted, application of the criteria will be considered. The total number of bobbin indications at BVPS-2 is less than 1/8th of the number of indications compared to the BVPS-1 OSGs, and less than 1/3 of the total number of bobbin indications at a similar point in the operational history. Therefore, tube support plate ODSCC is not expected to be a significant contributor to postulated SLB leakage contribution for several outages. As the voltage based repair criteria per GL 95-05 is not implemented, all bobbin indications at TSP intersections are inspected at each using the +Pt coil. Indications confirmed by +Pt examination are repaired.

#### 4.7.2.7.2.5 Cold Leg Thinning

Cold leg thinning has not been reported in the BVPS-2 steam generators through the Cycle 10 inspection.

#### 4.7.2.7.2.6 AVB Wear

AVB wear growth rates for the BVPS-2 steam generators are nearly equal to the AVB wear growth rates observed for the BVPS-1 OSGs. No AVB wear indications exceeded the Technical Specification repair limit at the 2R10, 2R09, 2R08, 2R07, or 2R06 inspections. Should AVB wear growth rates be significantly affected, the potential impact to tube integrity will be evaluated in the condition monitoring/operational assessment report. For the Cycle 10 operating period, the AVB wear growth rate at a 95% confidence level was [ ]$^{a,c}$ % per EFPY. Adjusting for a typical 500 EFPD operating period, the 95% confidence AVB wear growth rate is [ ]$^{a,c}$ % per EFPY. The low BVPS-2 AVB wear growth inherently provides large margins at the end of cycle conditions against the structural limit. This margin should not be significantly reduced following operation at EPU conditions.

#### 4.7.2.7.2.7 Axial PWSCC at Dented TSP Intersections

At the 2R09 outage, two tubes were reported with axial PWSCC indications at dented locations. This was the first observance of this mechanism at BVPS-2. These indications were identified by bobbin, and confirmed by +Pt. Flaw lengths were contained within the TSP bounds. All similar bobbin indications were inspected with +Pt; there were no additional confirmations. A review of the bobbin coil data from previous outages indicates that the indications could be observed for at least 3 operating cycles. Based on the history review, and 2R09 +Pt inspection scope, it is unlikely that this mechanism will be a significant contributor to primary-to-secondary leakage potential in the next few outages. This data also suggests that these indications are most likely representative of an outlier condition as opposed to a systematic condition within the BVPS-2 steam generators. No PWSCC at dented TSP intersections was reported at the 2R10 outage.

#### 4.7.2.7.2.8 Axial ODSCC at Freespan Dings

At the 2R09 outage, one tube was reported with an axial ODSCC indication at a freespan ding. This was the first observance of this mechanism at BVPS-2. The ding amplitude was 14 volts by bobbin. BVPS-2 uses the bobbin screening technique for detection of axial ODSCC in freespan dings originally developed and qualified for another plant with a several cycle history of axial ODSCC in freespan dings. The bobbin technique is qualified for detection of axial ODSCC up to and including 5 volt dings. At 2R09, the initial +Pt scope was 20% of freespan dings >5 volts. The +Pt scope was increased to 100% of the >5 volt freespan dings in the SG with the indication. No additional indications were reported. Due to the ding amplitude, bobbin history review will not provide additional information related to the initiation point of this indication. Both the inspection history from the other plants with ding ODSCC and the laboratory program that generated the flaw samples used for development of enhanced eddy current techniques indicate that the SCC does not extend past the stress field of the ding. As dings are typically <1/4" in length, ding ODSCC does not represent a significant contributor to primary-to-secondary leakage potential at EPU conditions. The growth data developed for the other plants with this mechanism (621°F operating temperature) indicate that the mechanism is manageable, and generally not a challenge to structural or leakage integrity. Growth rates at BVPS-2 should be significantly reduced compared to this other plant experience. No ding ODSCC indications were reported at the 2R10 outage.

#### 4.7.2.7.3 Results and Conclusions

Operational performance of the BVPS-2 steam generators with regard to stress corrosion cracking (SCC) mechanisms has been good. Through 12.6 EFPY, the total steam generator tube plugging is 3.8%, with only 1.4% plugging attributed to SCC mechanisms.

The increase in operating temperatures (i.e., best estimate $T_{hot} \leq 611°F$) associated with EPU conditions has been used to estimate SCC growth rates for the observed mechanisms at BVPS-2. The temperature increase is estimated to have a negligible impact upon crack growth rates. Structural and leakage integrity has been established for all reported SCC indications to date using both analytic methods and in situ pressure testing. Structural and leakage integrity should continue to be demonstrated post-EPU. Observed growth rate changes will be considered in future operational assessments. Degradation patterns reported to date suggest that the susceptible population of tubes that could experience ODSCC at the top of tubesheet region is limited. Incidence of RPC confirmed degradation at TSP intersections has been negligible. The recent inspection results for BVPS-2 suggest that the ODSCC at TSP intersections has an extremely low initiation and growth rate.

Ameliorative measures performed prior to operation have shown to be effective in greatly reducing the initiation potential of PWSCC mechanisms at BVPS-2. This is established since no PWSCC mechanisms in small radius U-bends has been reported to date at BVPS-2, and only one axial PWSCC indication at the top of tubesheet has been reported.

AVB wear growth rates have been calculated for recent inspections, and shown to be constant or reduced with successive inspections. The AVB wear growth rates are sufficiently low such that any growth rate increase associated with the EPU should result in end of cycle wear depths well below the associated structural limit.

In conclusion, all observed degradation mechanisms at BVPS-2 have been shown to provide both structural and leakage integrity at end of cycle conditions, and should result in similar conditions post-EPU.

The results and conclusions of the analyses and evaluations performed for steam generator tube degradation for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-2.

#### 4.7.2.7.4 References

1. EPRI TR-107197, "Depth Based Structural Analysis Methods for SG Circumferential Indications," November 1997.

## 4.8 PRESSURIZER

### 4.8.1 Introduction

The functions of the pressurizer are to absorb any expansion or contraction of the primary reactor coolant due to changes in temperature and/or pressure and, in conjunction with the pressure control system components, to keep the RCS at the desired pressure. The first function is accomplished by keeping the pressurizer approximately half full of water and half full of steam at normal conditions, connecting the pressurizer to the RCS at the hot leg of one of the reactor coolant loops and allowing inflow to or outflow from the pressurizer as required. The second function is accomplished by keeping the temperature in the pressurizer at the water saturation temperature ($T_{sat}$) corresponding to the desired pressure. The temperature of the water and steam in the pressurizer can be raised by operating electric heaters at the bottom of the pressurizer and can be lowered by introducing relatively cool spray water into the steam space at the top of the pressurizer.

The components in the lower end of the pressurizer (such as the surge nozzle, lower head/heater well and support skirt) are affected by pressure and surges through the surge nozzle. The components in the upper end of the pressurizer (such as the spray nozzle, safety and relief nozzle, upper head/upper shell, manway and instrument nozzle) are affected by pressure, spray flow through the spray nozzle, and steam temperature differences.

The limiting operating conditions of the pressurizer occur when the RCS pressure is high and the RCS hot leg ($T_{hot}$) and cold leg ($T_{cold}$) temperatures are low. This maximizes the $\Delta T$ that is experienced by the pressurizer. Due to flow out of and into the pressurizer during various transients, the surge nozzle alternately sees water at the pressurizer temperature ($T_{sat}$) and water from the RCS hot leg at $T_{hot}$. If the RCS pressure is high (which means, correspondingly, that $T_{sat}$ is high) and $T_{hot}$ is low, then the surge nozzle will see maximum thermal gradients; and, thus experience the maximum thermal stress. Likewise, the spray nozzle and upper shell temperatures alternate between steam at $T_{sat}$ and spray water, which, for many transients, is at $T_{cold}$. Thus, if RCS pressure is high ($T_{sat}$ is high) and $T_{cold}$ is low, then the spray nozzle and upper shell will also experience the maximum thermal gradients and thermal stresses.

To support the EPU Project, an evaluation was performed to address the impact on the pressurizer. This evaluation is based on the range of NSSS operating parameters that support an NSSS power level of 2910 MWt.

### 4.8.2 Input Parameters and Assumptions

The major inputs used in this evaluation are the EPU PCWG parameters provided in Section 2.1.1 and the EPU NSSS design transients provided in Section 2.2.1.

The EPU PCWG parameters and EPU NSSS design transients are considered in the EPU evaluations. No other changes are considered to the pressure or thermal/hydraulic design parameters for the EPU Project.

Seismic analyses and non-pressure boundary component evaluations are considered to be unaffected by the EPU Project.

The pressurizer analysis for the EPU Project also addressed pressurizer insurge/outsurge transients (Reference 1), consistent with the analytical assumptions employed in the evaluation of pressurizer transients (Reference 2 for BVPS-1 and Reference 3 for BVPS-2).

### 4.8.3 Description of Analyses and Evaluations

The analysis for EPU was performed by evaluating the original pressurizer stress reports, which were performed to the requirements of the ASME Boiler and Pressure Vessel Code, Section III (1965 Edition, Winter 1966 Addendum (Reference 4) for BVPS-1 and 1971 Edition, Summer 1972 Addendum (Reference 5) for BVPS-2). Analytical models of various sections of the pressurizer were subjected to pressure loads, external loads (such as piping loads), and thermal transients.

The input parameters associated with the EPU Project were reviewed and compared to the design inputs considered in the original pressurizer stress reports. In cases where revised input parameters are not bounded, pressurizer structural analyses and evaluations were performed. Any impacts to the existing design basis analysis were performed through a comparative analysis of the changes. This method involves a simplified engineering approach, using the existing analyses as the basis of the evaluation. Scaling factors were utilized to assess the impact of the changes in the parameters such as the system transients, temperatures, and pressures. New stresses and revised cumulative usage factors were calculated, as applicable, and compared to previous licensed results. The evaluation results were then compared with the ASME Code (Reference 4 for BVPS-1 and Reference 5 for BVPS-2) to confirm that the allowable limits are maintained.

In addition, the impact of the EPU parameters and design transient changes on the insurge/outsurge transients was evaluated for the pressurizer lower head critical locations evaluated in References 2 and 3, based on the operating scenario that produced acceptable fatigue results. These results are tabulated separately from the results of the evaluation to assess the original stress report for EPU conditions as described above.

For BVPS-1, the impact of changes to the BVPS-1 spray nozzle transients was also evaluated for the critical spray nozzle location. These evaluations were based on the ASME Code Section III, 1989 Edition (Reference 6), consistent with and as justified in Reference 2.

### 4.8.4 Acceptance Criteria and Results

The current design inputs to the pressurizer stress reports were compared to the corresponding design inputs for the EPU Project.

If comparison of the design inputs revealed hot and/or cold leg temperatures, NSSS design transients, or design loads that did not comply with the current design inputs, pressurizer structural analyses and evaluations were performed, as necessary, to incorporate the revised design inputs. The acceptance criterion is that the pressurizer components meet the stress/fatigue analysis requirements of the ASME Code, Section III (References 4 and 6 for BVPS-1 and References 5 and 6 for BVPS-2) for plant operation in accordance with the EPU Project.

The critical pressurizer components are the spray nozzle, upper head, surge nozzle, lower head, heater well, support skirt and flange, safety and relief nozzles, instrument nozzle, immersion heater, and seismic support lug. For BVPS-2, the shell buildup at trunnion is also a critical component. The results of the EPU evaluation are described below. The fatigue usage summary for the pressurizer is presented in Tables 4.8-1 (not considering insurge/outsurge operating transients) and 4.8-3 (considering insurge/outsurge operating transients). The primary plus secondary stress intensity ranges are presented in Table 4.8-2 (not considering insurge/outsurge operating transients).

**Spray Nozzle**

For BVPS-1, summary stress results for the spray nozzle for the original power rating were evaluated to assess the changes that occur due to the EPU conditions. Design, faulted and test condition stresses remain unchanged. Results for the primary plus secondary stress intensity range and fatigue usage for normal and upset conditions were evaluated. Scale factors were developed to account for those transients outside the original design envelope, and the fatigue usage recalculated at the most critical location. The ASME Code requirements are satisfied.

For BVPS-1, a supplemental evaluation had been performed to address the effect of out-of-specification transients on the spray nozzle. Since the fatigue usage calculated is less than that previously determined, it was conservative to base the EPU evaluation on the previous value.

For BVPS-2, the EPU transients were compared to the transients used for the normal power condition and found to be bounded by the original design transients. Therefore the existing results remain valid. The ASME Code requirements are satisfied.

**Upper Head**

For BVPS-1 and BVPS-2, summary stress results for the upper head for the original power rating were evaluated to assess the changes that occur due to the EPU conditions. Design, faulted and test condition stresses remain unchanged. Results for the primary plus secondary stress intensity range and fatigue usage for normal and upset conditions were evaluated. Scale factors were developed to account for those transients outside the original design envelope, and the fatigue usage recalculated at the most critical location. The ASME Code requirements are satisfied.

**Surge Nozzle**

For BVPS-1, the EPU transients were compared to the transients used for the original power condition and found to be bounded by the original design transients. Therefore the existing results remain valid. The ASME Code requirements are satisfied.

For BVPS-2, summary stress results for the surge nozzle for the original power rating were evaluated to assess the changes that occur due to the EPU conditions. Design, faulted and test condition stresses remain unchanged. Results for the primary plus secondary stress intensity range and fatigue usage for normal and upset conditions were evaluated. Those transients outside the original design envelope were moved into transient groups with higher ΔT's, and the fatigue usage recalculated at the most critical location. The ASME Code requirements are satisfied.

For BVPS-1 and BVPS-2, the EPU transients were also compared to the insurge/outsurge operational transients of References 2 and 3. Significant changes to the insurge/outsurge transients were re-evaluated for the most limiting location in the pressurizer surge nozzle. For BVPS-1, existing results from Reference 2 remain valid. For BVPS-2, the maximum fatigue usage was determined to be [ ]$^{a,c}$ at the safe end to pipe weld. This is reflected in Table 4.8-3.

**Lower Head**

For BVPS-1, summary stress results for the lower head for the original power rating were evaluated to assess the changes that occur due to the EPU conditions. Design, faulted and test condition stresses remain unchanged. Results for the primary plus secondary stress intensity range and fatigue usage for normal and upset conditions were evaluated. Scale factors were developed to account for those transients outside the original design envelope, and the fatigue usage recalculated at the most critical location. The ASME Code requirements are satisfied.

For BVPS-2, the EPU transients were compared to the transients used for the normal power condition and found to be bounded by the original design transients. Therefore the existing results remain valid. The ASME Code requirements are satisfied.

For BVPS-1 and BVPS-2, the EPU transients were also compared to the insurge/outsurge operational transients of References 2 and 3. Significant changes to the insurge/outsurge transients were re-evaluated for the most limiting location in the pressurizer lower head (heater penetration). For BVPS-1, existing results from Reference 2 remain valid. For BVPS-2, the maximum fatigue usage was determined to be [ ]$^{a,c}$ in the lower head at the heater penetration. This is reflected in Table 4.8-3.

**Heater Well**

For BVPS-1, summary stress results for the heater well for the original power rating demonstrated that the only transient with a significant thermal effect on the stresses at the heater well was Reactor Trip, and that the thermal contribution from other transients may be neglected. Therefore, since the ΔT in the original stress report envelopes the EPU ΔT for the Reactor Trip transient, the original design transients bound the EPU transients and the existing results remain valid. The ASME Code requirements are satisfied.

For BVSP-2, the EPU transients were compared to the transients used for the normal power condition and found to be bounded by the original design transients. Therefore the existing results remain valid. The ASME Code requirements are satisfied.

**Support Skirt and Flange**

For BVPS-1, the EPU transients were compared to the transients used for the original power condition and found to be bounded by the original design transients. Therefore, the existing results remain valid. The ASME Code requirements are satisfied.

For BVPS-2, the thermal transients within the pressurizer do not contribute to the fatigue usage of the support skirt and flange. Therefore, the original design transients bound the EPU transients and the existing results remain valid. The ASME Code requirements are satisfied.

**Safety and Relief Nozzles**

Summary stress results for the safety and relief nozzle for the original power rating were evaluated to assess the changes that occur due to the EPU conditions. Design, faulted and test condition stresses remain unchanged. Results for the primary plus secondary stress intensity range and fatigue usage for normal and upset conditions were evaluated.

For BVPS-1, scale factors were developed to account for those transients outside the original design envelope, and the fatigue usage recalculated at the most critical location. The maximum fatigue usage was determined to be [ ]$^{a,c}$. The ASME Code requirements are satisfied.

For BVPS-2, a combination of shifting transients into transient groups with higher ΔT's and developing scale factors was used to account for those transients outside the original design envelope, and the fatigue usage was recalculated at the most critical location. The ASME Code requirements are satisfied.

**Instrument Nozzle**

For BVPS-1, the EPU transients were compared to the transients used for the original power condition and found to be bounded by the original design transients. Therefore the existing results remain valid. The ASME Code requirements are satisfied.

For BVPS-2, summary stress results for the instrument nozzle for the original power rating were evaluated to assess the changes that occur due to the EPU conditions. Design, faulted and test condition stresses remain unchanged. Results for the primary plus secondary stress intensity range and fatigue usage for normal and upset conditions were evaluated. Those transients outside the original design envelope were moved into transient groups with higher ΔT's, and the fatigue usage recalculated at the most critical location. The ASME Code requirements are satisfied.

**Immersion Heater**

For BVPS-1, the EPU transients were compared to the transients used for the original power condition and found to be bounded by the original design transients. Therefore the existing results remain valid. The ASME Code requirements are satisfied.

For BVPS-2, the stress report demonstrated that the maximum alternating stress at any of the critical locations was below the endurance limit of the material for the most severe thermal transient. Therefore the EPU thermal transients have no effect on either the heater sheath or the heater well weld and the existing results remain valid. The ASME Code requirements are satisfied.

**Seismic Support Lug**

For BVPS-1 and BVPS-2, only seismic loads contribute to fatigue of the support lug. Therefore the EPU thermal transients have no effect on fatigue of the support lug and the existing results remain valid. The ASME Code requirements are satisfied.

**Shell Buildup at Trunnion (BVPS-2 Only)**

For BVPS-2, summary stress results for the shell buildup at the trunnion for the original power rating were modified to reflect the changes that occur due to the EPU conditions. Design, faulted and test condition stresses remain unchanged. Results for the primary plus secondary stress intensity range and fatigue usage for normal and upset conditions were evaluated. Scale factors were developed to account for those transients outside the original design envelope, and the fatigue usage recalculated at the most critical location. The ASME Code requirements are satisfied.

### 4.8.5 Conclusions

The BVPS-1 and BVPS-2 pressurizer fatigue usage factors after EPU are given in Table 4.8-1. The BVPS-1 and BVPS-2 primary plus secondary stress intensity ranges after EPU are given in Table 4.8-2. These results update the results of the original pressurizer stress reports and do not consider insurge/outsurge operating transients. Table 4.8-3 shows the fatigue usage for the limiting pressurizer lower head and surge nozzle locations considering insurge/outsurge operating transients after EPU.

For BVPS-1 and BVPS-2, all of the critical components for the pressurizer have been evaluated for operation at the EPU conditions. It was determined that all components continue to satisfy the applicable ASME Code requirements. As such, the pressurizer is qualified for operation at the plant conditions defined for the EPU Project.

The results and conclusions of the analyses and evaluations performed for the pressurizer for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

### 4.8.6 References


1. WCAP-14950, "Mitigation and Evaluation of Pressurizer Insurge/Outsurge Transients," February 1998.

2. WCAP-15351, "Evaluation of Pressurizer Transients Based on Plant Operations for Beaver Valley Unit 1," March 2000.

3. WCAP-15352, "Evaluation of Pressurizer Transients Based on Plant Operations for Beaver Valley Unit 2," March 2000.

4. ASME Boiler and Pressure Vessel Code, Section III, 1965 Edition with Addenda through Winter 1966, American Society of Mechanical Engineers, New York, New York.

5. ASME Boiler and Pressure Vessel Code, Section III, 1971 Edition with Addenda through Summer 1972, American Society of Mechanical Engineers, New York, New York.

6. ASME Boiler and Pressure Vessel Code, Section III, 1989 Edition, American Society of Mechanical Engineers, New York, New York.

| Table 4.8-1<br>Pressurizer Fatigue Usage<br>Not Considering Insurge/Outsurge Operating Transients | | |
|---|---|---|
| **Component** | **BVPS-1 Fatigue Usage** | **BVPS-2 Fatigue Usage** |
| Spray Nozzle | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Upper Head | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Surge Nozzle | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Safety and Relief Nozzles | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Support Skirt and Flange | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Lower Head | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Heater Well | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Seismic Support Lug | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Shell Buildup at Trunnion | N/A(1) | [ ]$^{a,c}$ |
| Instrument Nozzle | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Immersion Heater | [ ]$^{a,c}$ - | [ ]$^{a,c}$ (2) |

Notes

(1) This location is not limiting.

(2) The stresses are below the endurance limit, so the fatigue life is essentially infinite based on the ASME fatigue curves and the fatigue usage is zero.

| Table 4.8-2<br>Pressurizer Primary Plus Secondary Stress Intensity Ranges<br>Not Considering Insurge/Outsurge Operating Transients | | |
|---|---|---|
| **Component** | **BVPS-1<br>Stress Intensity Ratio(1)** | **BVPS-2<br>Stress Intensity Ratio(1)** |
| Spray Nozzle | [ ]$^{a,c\,(2)}$ | [ ]$^{a,c\,(2)}$ |
| Upper Head | [ ]$^{a,c}$ | [ ]$^{a,c\,(2)}$ |
| Surge Nozzle | [ ]$^{a,c}$ | [ ]$^{a,c\,(2)}$ |
| Safety and Relief Nozzles | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Support Skirt and Flange | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Lower Head | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Heater Well | [ ]$^{a,c}$ | [ ]$^{a,c\,(2)}$ |
| Seismic Support Lug | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Shell Buildup at Trunnion | N/A(3) | [ ]$^{a,c\,(2)}$ |
| Instrument Nozzle | [ ]$^{a,c}$ | [ ]$^{a,c\,(2)}$ |
| Immersion Heater | [ ]$^{a,c}$ | [ ]$^{a,c}$ |

Notes:

(1) Ratio of calculated to allowable stress intensity.

(2) These values that exceed the $3S_m$ limit on the range of primary plus secondary stress intensity are justified by simplified elastic-plastic analysis in accordance with Paragraph NB-3228.3 in ASME III, Subsection NB.

(3) This location is not limiting.

| Table 4.8-3<br>Pressurizer Fatigue Usage<br>Considering Insurge/Outsurge Operating Transients | | |
|---|---|---|
| **Limiting Location** | **BVPS-1 Fatigue Usage** | **BVPS-2 Fatigue Usage** |
| Surge Nozzle Safe End to Pipe Weld | [ ]$^{a,c}$ | [ ]$^{a,c}$ |
| Lower Head at Heater Penetration | [ ]$^{a,c}$ | [ ]$^{a,c}$ |

## 4.10 LOOP STOP ISOLATION VALVES

### 4.10.1 Introduction

This section addresses the ASME Code structural considerations for the pressure boundary components of the Loop Stop Isolation Valves (LSIVs). The LSIVs were evaluated for the EPU Project PCWG parameters and the associated NSSS design transients.

This evaluation provides verification of continued structural suitability of the pressure boundary components of the existing LSIVs for the EPU Project.

### 4.10.2 Input Parameters and Assumptions

The BVPS-1 LSIVs were originally designed and analyzed to the ASME Code generic component design reports. The individual components were designed to various addenda of the 1968 ASME Code referenced in the original component reports. If no specific code addenda were listed then the one applicable for the valve body is applied. The BVPS-2 LSIVs were originally designed and analyzed to the ASME Code 1971 Edition through Winter of 1973 Addenda including Code Case 1552.

The input parameters that were used to perform the analyses and evaluations for EPU include the original NSSS design parameters and NSSS design transients, the EPU PCWG parameters (Section 2.1.1) along with the EPU NSSS design transients (Section 2.2.1), and the current design basis evaluations for the LSIVs.

### 4.10.3 Description of Analyses and Evaluations

The evaluation consists of a review of the original component reports for the valve, the evaluation performed for the 1.4% uprating, and evaluations of the LSIVs resulting from other plant upratings to determine if the conditions considered in previous evaluations envelop the EPU conditions. For components that have conditions not enveloped by the original analysis or previous evaluations, a supplemental analysis is performed.

The design inputs that change as a result of the EPU are the operating temperature, and the pressure and temperature differences during the design transients. The design transients were reviewed to evaluate pressure and temperature changes due to EPU.

Two items must be addressed to verify the valve is acceptable for the conditions defined for the EPU conditions. The first is to evaluate if the revised operating conditions are enveloped by previous analysis (i.e., the temperatures used in the previous analysis envelop those for the EPU), and second is to evaluate the effect of the EPU transients on fatigue for the pressure boundary components.

#### 4.10.3.1 BVPS-1 LSIV Component Evaluations

**Valve Body (1968 Code through Winter of 1968 Addenda)**

A review of the previous evaluation for the LSIVs at BVPS-1 showed that the body was evaluated for an operating temperature of 650°F. Therefore, the main body is acceptable for the revised maximum operating temperature of 617°F.

The effect of the transients for the EPU on the fatigue usage factors for the body was evaluated. Since the revised design transients for the EPU are not significantly different from the original conditions, the revised conditions are evaluated to determine if the fatigue waiver is still applicable to the revised transients. The only significant difference is the COMS pressure and temperature transients.

The COMS pressure cycling is addressed by the maximum allowable pressure variation. The maximum allowable pressure is greater than the pressure change for COMS. Therefore, the COMS pressure cycles are enveloped by the previous waiver.

The other condition is the COMS temperature transients. The evaluation showed that the fatigue waiver is still applicable to these temperature transients.

Based on the original criteria and the revised transients, the body is exempt from fatigue evaluation.

**Bonnet (1968 Code)**

The bonnet is evaluated at a design temperature of 650°F. Therefore, the revised operating condition, with a maximum temperature of 617°F, is enveloped by the original analysis.

The revised design transients were reviewed to determine the effect on fatigue usage. The total fatigue usage factor for the bonnet is [ ]$^{a,c}$.

**Main Flange Bolting (1968 Code through Summer 1968 Addenda)**

The main flange bolting is evaluated at a design temperature of 650°F. Therefore, the revised operating condition, with a maximum temperature of 617°F, is enveloped by the original analysis.

The revised design transients were reviewed to determine the effect on fatigue usage. The total fatigue usage factor for the main flange bolting is [ ]$^{a,c}$.

**Disc**

The disc is analyzed for design conditions of 2500 psia at 650°F, which envelops the revised maximum operating temperature of 617°F. Therefore, the valve is acceptable for the revised operating conditions.

The revised design transients were reviewed to determine the effect on fatigue usage. The total fatigue usage factor for the disc is [ ]$^{a,c}$.

**Stem and Yoke**

The stem is analyzed to envelop the design conditions of 2500 psia at 650°F, which envelops the revised maximum operating temperature of 617°F. Therefore, the valve is acceptable for the revised operating conditions.

Considering the revised design transients and the resulting transient groupings, the fatigue usage factor remains acceptable.

**Canopy Seal Ring (1968 Edition through Winter 1968 Addenda)**

The Canopy Seal Ring is analyzed to envelop the design conditions of 2500 psia at 650°F, which envelops the revised maximum operating temperature of 617°F. Thus, the valve is acceptable for the revised operating conditions.

The revised design transients were reviewed to determine the effect on fatigue usage. The revised design transients and the resulting transient groupings are not significantly different than the original design transients which result in a fatigue usage factor of [ ]a,c, thus the fatigue usage factor remains acceptable.

**Backseat**

The backseat is designed for a pressure of 2500 psia at 650°F. Therefore, the design would envelope the revised maximum operating temperature of 617°F, and the valve is acceptable for the revised operating conditions.

The revised design transients were reviewed to determine the effect on fatigue usage. The revised design transients and the resulting transient groupings are not significantly different than the original design transients which result in a fatigue usage factor of [ ]a,c, thus the fatigue usage factor remains acceptable.

**Other Components**

In a similar manner, other LSIV components were evaluated and the resulting fatigue usage factors were shown to be acceptable.

### 4.10.3.2 BVPS-2 LSIV Component Evaluations

**Valve Body**

A review of the current design basis evaluation for the LSIVs at BVPS-2 showed that the body was evaluated for a normal operating temperature of 618°F. Further, the original body design analysis was reviewed to confirm this temperature. Based on the original analysis, a temperature of 619°F was used for the normal operating conditions for the blending region of the main flange. This same temperature was used for the normal hot leg temperatures in the upset and emergency cases. Therefore, the lowest temperature was determined to be 618°F, based on a review of the normal, upset, emergency and faulted conditions.

Based on the review, the minimum operating temperature for all four operating conditions was 618°F which is greater than the 617°F operating temperature for the EPU Project. Therefore, the main body is acceptable for the new operating temperature.

The revised design transients for the EPU Project were reviewed to determine the effect on the fatigue usage factors for the body. The body weld end and main body blending region required re-evaluation.

The valve cyclic stresses are related to the rate of change in temperature and the severity of the temperature and pressure change. For the transients considered in the EPU Project, the change in temperature is greater than the change in pressure. Therefore, for all cases where the temperature change is greater for the new conditions, the stresses were ratioed by the change in temperature between the new conditions and the original conditions.

**Body Weld End**

The revised design transients were reviewed to determine the effect on fatigue usage. The total fatigue usage factor for the body weld end is [ ][a,c].

**Main Body Blending Region**

The revised design transients were reviewed to determine the effect on fatigue usage. The total fatigue usage factor for the main body blending region is [ ][a,c].

**Bypass Nozzle**

The revised design transients were reviewed to determine the effect on fatigue usage. The total fatigue usage factor for the bypass nozzle is [ ][a,c].

**Bonnet**

For the bonnet, two areas were evaluated for the revised design transients. They were the bonnet flange region and the bonnet shell flange juncture region. The total fatigue usage factors are [ ][a,c] and [ ][a,c], respectively.

**Other Components**

In similar manner, other LSIV components including the main flange bolting were evaluated and the resulting fatigue usage factors were shown to be acceptable. The total fatigue usage factor for the main flange bolting is [ ][a,c].

### 4.10.4 Acceptance Criteria and Results

The BVPS-1 and BVPS-2 LSIVs were evaluated for the EPU PCWG parameters and the associated NSSS design transients. In all cases, the existing analyses and evaluations remained applicable and bounding for the Design Temperature and Pressure.

New calculations were performed for fatigue usage and showed acceptable results for all LSIV components as described in Section 4.10.3.

The limiting LSIV component for BVPS-1 is the main flange bolting with a total fatigue usage factor of [ ]$^{a,c}$. The limiting LSIV component for BVPS-2 is the main flange bolting with a total fatigue usage factor of [ ]$^{a,c}$. These total fatigue usage factors are acceptable with respect to the acceptance criteria of 1.0.

### 4.10.5 Conclusions

The BVPS-1 and BVPS-2 LSIVs were evaluated for the EPU PCWG parameters (Section 2.1.1) and the associated NSSS design transients (Section 2.2.1). In all cases, the existing analyses and evaluations remained applicable and bounding for the Design Temperature and Pressure. New calculations were performed for fatigue usage to establish the structural acceptability of the LSIV pressure boundary components in accordance with the ASME Code.

Based on the previous analyses and the analyses and evaluations performed for EPU, the BVPS-1 and BVPS-2 LSIV pressure boundary components are acceptable in accordance with the ASME Code for the EPU Project.

The results and conclusions of the analyses and evaluations performed for the loop stop isolation valves for the NSSS power of 2910 MWt bound and support operation at the current NSSS power of 2697 MWt, thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

# 5 SAFETY ANALYSIS

The EPU Project included safety (accident) analyses for the Updated Final Safety Analysis Report (UFSAR) transients and accidents at EPU conditions. This section includes the evaluation of initial condition uncertainties at EPU conditions which are provided as input to the safety (accident) analyses and the development of any changes to reactor trip system (RTS)/engineered safety feature actuation system (ESFAS) setpoints as a result of the safety (accident) analyses.

In addition to initial condition uncertainties and RTS/ESFAS setpoint changes, the following safety (accident) analyses at EPU conditions are also addressed in this section:

- LOCA transients
- Non-LOCA transients
- Steam Generator Tube Rupture
- LOCA Mass and Energy Releases
- MSLB Mass and Energy Releases
- LOCA Hydraulic Forces
- Anticipated Transients Without Scram
- Natural Circulation and Cooldown
- Radiological Dose Consequences
- Fire Protection (Appendix R) Safe Shutdown

The analyses and evaluations presented in this section support operation of BVPS-1 at EPU conditions with the Model 54F replacement steam generators (RSGs) and BVPS-2 at EPU conditions with the original steam generators. The analyses and evaluations for EPU conditions bound and support operation at the current power level, which supports the staged implementation of EPU at BVPS-1 and BVPS-2.

For BVPS-1, the safety (accident) analyses at EPU conditions support elimination of the boric acid concentration requirement for the boron injection tank.

## 5.1 INITIAL CONDITION UNCERTAINTIES

### 5.1.1 Introduction

Initial condition uncertainties are conservative steady state instrumentation measurement uncertainties that are applied to nominal parameter values in order to obtain conservative initial conditions for use in safety (accident) analyses. The initial condition uncertainties were recalculated at EPU conditions for use in the EPU Project analyses and/or evaluations to assess the acceptability of the safety analyses at EPU conditions. The initial condition uncertainties for EPU conditions were provided as input to the loss-of-coolant accident (LOCA) analysis (Section 5.2), non-LOCA analysis (Section 5.3), steam generator tube rupture (SGTR) analysis (Section 5.4), LOCA mass and energy release analysis (Section 5.5), main steamline break mass and energy release analysis (Section 5.6), LOCA hydraulic forces analysis (Section 5.7), and fuel thermal-hydraulic design analysis (Section 6.1).

### 5.1.2 Input Parameters and Assumptions

The uncertainty calculations for the Beaver Valley Power Station were performed based on the plant-specific instrumentation and plant calibration and calorimetric procedures.

### 5.1.3 Description of Analyses and Evaluations

The uncertainty analysis uses the Square-Root-Sum-of-the-Squares (SRSS) technique to combine the uncertainty components of an instrument channel in an appropriate combination of those components, or groups of components, which are statistically independent. Those uncertainties that are not independent are arithmetically summed to produce groups that are independent of each other, which can then be statistically combined. The methodology used for the EPU conditions is defined in Reference 1 and Reference 2 for BVPS-1 and BVPS-2, respectively, and is the same as was used for the NRC approved 1.4% measurement uncertainty recapture power uprate.

Initial condition uncertainties were calculated for the following six parameters that are explicitly modeled in the Beaver Valley Power Station safety analyses:

- Pressurizer Pressure Control - Automatic pressurizer pressure control system
- RCS $T_{avg}$ Control - Automatic reactor control system
- Reactor Power - Daily calorimetric power measurement [Rated Thermal Power (RTP)] used to normalize power range instruments
- RCS Total Flow - Loop RCS flow measurements based on RCS loop flow channels normalized to a once per fuel cycle calorimetric RCS flow measurement to verify Thermal Design Flow (TDF)
- Steam Generator Water Level Control - Automatic steam generator water level control system
- Pressurizer Water Level Control - Automatic pressurizer water level control system

In order to support the start of analyses and/or evaluations for safety analyses early in the EPU Project, preliminary initial condition uncertainties for EPU were provided as input to safety analyses and/or evaluations. The initial condition uncertainties for EPU were then calculated and finalized at the end of the project at which time it was confirmed that the final values were bounded by the preliminary values, except for BVPS-2 steam generator water level control where the initial condition uncertainty increased due to the resolution of generic level control uncertainty issues (References 3 and 4) unrelated to EPU. The safety analyses for EPU include the resolution of the generic steam generator water level control uncertainty issues (References 3 and 4). Although various safety analyses and/or evaluations for EPU might incorporate the preliminary initial condition uncertainties that differ from the calculated final values, the preliminary initial condition uncertainties include margin relative to the calculated final values.

### 5.1.4 Acceptance Criteria and Results

There are no explicit acceptance criteria for the initial condition uncertainties; however, the associated safety (accident) analyses must satisfy their applicable acceptance criteria. The initial condition uncertainties for safety analyses are documented in the UFSAR (Chapter 14 for BVPS-1 and Chapter 15 for BVPS-2). Once defined and incorporated into the safety analyses, the calculated final initial condition uncertainties must be less than or equal to the initial condition uncertainty values used in the safety analyses.

The results of the initial condition uncertainty analysis for EPU are summarized in Table 5.1-1A and Table 5.1-1B for BVPS-1 and BVPS-2, respectively. The results for pressurizer pressure control, RCS $T_{avg}$ control, reactor power and RCS total flow are documented in Reference 1 and Reference 2 for BVPS-1 and BVPS-2, respectively.

### 5.1.5 Conclusions

Preliminary initial condition uncertainties were determined for EPU conditions and were provided as input to the EPU Project safety analyses and/or evaluations. Final initial condition uncertainties were calculated at the end of the project at which time it was confirmed that the final values were bounded by the preliminary initial condition uncertainties, except for BVPS-2 steam generator water level control where the initial condition uncertainty increased due to the resolution of generic level control uncertainty issues (References 3 and 4) unrelated to EPU. The safety analyses for EPU include the resolution of the generic steam generator water level control uncertainty issues (References 3 and 4).

The results and conclusions of the analyses and evaluations performed for initial condition uncertainties for the reactor power of 2900 MWt (2910 MWt NSSS power) bound and support operation at the current reactor power of 2689 MWt (2697 MWt NSSS power), thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

### 5.1.6 References


1. WCAP-15264, Rev. 4, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology Beaver Valley Power Station Unit 1," October 2002.

2. WCAP-15265, Rev. 4, "Westinghouse Revised Thermal Design Procedure Instrument Uncertainty Methodology Beaver Valley Power Station Unit 2," October 2002.

3. NSAL-03-9, "Steam Generator Water Level Uncertainties," September 22, 2003.

4. TB-04-12, "Steam Generator Level Process Pressure Evaluation," June 23, 2004.

**Table 5.1-1A**
**BVPS-1 Summary of Initial Condition Uncertainties**

| Parameter | Preliminary Initial Condition Uncertainties[1] | Calculated Final Initial Condition Uncertainties[1] |
|---|---|---|
| Pressurizer Pressure Control | | a,c |
| RCS $T_{avg}$ Control | | |
| Reactor Power | | |
| RCS Total Flow | | |
| Steam Generator Water Level Control (@ 65% NRS)[2] | | |
| Pressurizer Water Level Control | | |

Notes:

(1) A negative bias means the channel indicates lower than actual and a positive bias means the channel indicates higher than actual.

(2) The calculated final initial condition uncertainty for steam generator water level control is calculated consistent with the recommendations in Nuclear Safety Advisory Letter NSAL-03-9 (Reference 3) and Technical Bulletin TB-04-12 (Reference 4).

| Table 5.1-1B<br>BVPS-2 Summary of Initial Condition Uncertainties | | |
|---|---|---|
| **Parameter** | **Preliminary Initial Condition Uncertainties(1)** | **Calculated Final Initial Condition Uncertainties(1)** |
| | | a,c |
| Pressurizer Pressure Control | | |
| RCS $T_{avg}$ Control | | |
| Reactor Power | | |
| RCS Total Flow | | |
| Steam Generator Water Level Control (@ 44% NRS)(2) | | |
| Pressurizer Water Level Control | | |

Notes:

(1) A negative bias means the channel indicates lower than actual and a positive bias means the channel indicates higher than actual.

(2) The calculated final initial condition uncertainty for steam generator water level control is calculated consistent with the recommendations in Nuclear Safety Advisory Letter NSAL-03-9 (Reference 3) and Technical Bulletin TB-04-12 (Reference 4).

# 6 FUEL ANALYSIS

This section describes the analyses and evaluations performed in the nuclear fuel and fuel-related areas to support the EPU Project. The section addresses analyses and evaluations performed for fuel thermal-hydraulic design, fuel nuclear core design, fuel rod design and performance, heat generation rates, and neutron fluence.

The analyses and evaluations presented in this section support operation of BVPS-1 at EPU conditions with the Model 54F replacement steam generators (RSGs) and BVPS-2 at EPU conditions with the original steam generators. The analyses and evaluations for EPU conditions bound and support operation at the current power level, which supports the staged implementation of EPU at BVPS-1 and BVPS-2.

The analyses and evaluations for the nuclear fuel and fuel-related areas at EPU conditions use a total core peaking factor ($F_Q$) of 2.4 or 2.52 and a nuclear enthalpy rise hot channel factor ($F_{\Delta H}$) of 1.62 or 1.75. In all cases, the nuclear fuel and fuel-related analyses and evaluations for EPU conditions support a minimum $F_Q$ of 2.4 and a minimum $F_{\Delta H}$ of 1.62. The use of larger peaking factors in select analyses and evaluations supports the potential for a future increase in peaking factors at EPU conditions.

## Fuel Assembly Design

To support EPU, the fuel assembly design for BVPS was changed from the 17x17 VANTAGE 5H/PERFORMANCE+ (w/o Intermediate Flow Mixing (IFM) grids) design to the 17x17 Robust Fuel Assembly (RFA) design (w/ IFMs), including the RFA-2 design. The RFA-2 design is essentially identical to the RFA design except that it includes an enhanced mid grid design that results in increased mid grid contact area with the fuel rod. The enhanced mid grid design has no impact on the fuel assembly thermal hydraulic, neutronics, or structural models. The RFA design contains a mid grid allowable structural limitation that is conservative with respect to the RFA-2 design. The analyses and evaluations performed for RFA fuel also apply to RFA-2 fuel, and the term RFA fuel as used in this report includes applicability to RFA-2 fuel.

The transition to RFA fuel was initiated at the current core power level (2689 MWt). It is anticipated that the fuel transition will be complete and the entire core will consist of RFA fuel when EPU is implemented. Although the core will be fully transitioned to RFA fuel when EPU is implemented, previously burned VANTAGE 5H fuel assemblies may be reinserted into the core as part of a cycle specific reload. The VANTAGE 5H fuel design is mechanically and hydraulically compatible with the RFA fuel design. The acceptability of reinserting VANTAGE 5H fuel assemblies into the core will be confirmed during the normal reload design process for the specific loading pattern chosen for that reload design.

A description of the RFA and V5H fuel assembly mechanical design features is provided in this section.

## Fuel Mechanical Design Features

This section describes the mechanical design and the compatibility of the 17x17 RFA fuel assembly design (w/ IFMs) and the VANTAGE 5H/PERFORMANCE+ (w/o IFMs) fuel assembly design. The RFA fuel assembly is designed to be compatible with the VANTAGE 5H fuel assembly, reactor internals interfaces, the fuel handling equipment and refueling equipment. The RFA design dimensions are essentially equivalent to the VANTAGE 5H assembly design from an exterior assembly envelope and reactor internals interface standpoint.

The significant mechanical features of the RFA design that differ from the VANTAGE 5H design are the addition of three IFM grids, modification to the mixing vane mid grids, and increased thimble and instrument tube outer diameters. Details of the RFA fuel assembly design are presented in the following sections.

## Design Description of the 17x17 Robust Fuel Assembly

The 17x17 RFA design is a 17x17 array with the standard fuel rod design 0.374 in. rod outside diameter. The design incorporates and adapts many of the current Westinghouse advanced fuel features, including:

- ZIRLO™ thick thimble and instrument tubes
- Removable Top Nozzle (RTN)
- Reduced Rod Bow (RRB) Inconel Top Grid
- ZIRLO™ Modified Low Pressure Drop (LPD) Structural Mid Grids
- ZIRLO™ Modified Intermediate Flow Mixing Grids
- High Burnup Inconel Bottom Grid
- Debris Filter Bottom Nozzle
- Inconel Protective Bottom Grid
- Zirconium oxide coating on the bottom section of the fuel rod
- Debris mitigating long fuel rod bottom end plugs
- ZIRLO™ Clad Fuel Rods

The 17x17 RFA design is a VANTAGE 5H/PERFORMANCE+ design (STD fuel rod size of 0.374 in. outer rod diameter) using LPD structural and IFM grids of a modified design.

The RFA design incorporates three ZIRLO™ IFM grids. The RFA design is mechanically and hydraulically compatible with the VANTAGE 5H (w/o IFMs), and the same functional requirements and design criteria apply to the Westinghouse RFA fuel assemblies and VANTAGE 5H (w/o IFMs) fuel assemblies.

## Fuel Rods

The RFA fuel rod has the same clad wall thickness, fuel rod pellet stack active length, fuel rod diameter, bottom end plug and cladding material (ZIRLO™) as the VANTAGE 5H fuel rod.

## Grid Assemblies

The RFA fuel includes IFM grids. The IFM's primary function is to promote flow mixing. Additionally, they limit rod bow in the hottest fuel assembly spans. They must accomplish this without inducing clad wear beyond established limits. The IFMs must avoid interactive damage with grids from neighboring fuel assemblies during core loading or unloading operations.

The IFM grids are located in the three uppermost spans between the ZIRLO™ mixing vane structural grids and incorporate a similar mixing vane array. Their prime function is mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with the mixing vanes. This simplified cell arrangement allows short grid cells so that the IFM can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids and mixing vane grids are fabricated from ZIRLO™. This material was selected to take advantage of the material's inherent low neutron capture cross section.

The RFA mid grid has a mixing vane pattern. Differences between the RFA mid grids and IFM grids, and the VANTAGE 5H fuel include:

- Mixing vane pattern
- Vane geometry
- Spring and dimple geometry
- Intersect slot length

To allow for the larger thimble and instrument tubes, the RFA mid grids and IFM grids are embossed (radiused) at the thimble cell locations to accept the larger diameter thimble tube.

The Inconel bottom and protective bottom grids are the same for RFA and VANTAGE 5H except for the larger insert inner diameter.

Guide Thimble and Instrument Tubes

The RFA design incorporates thicker walled thimble and instrumentation tubes relative to the VANTAGE 5H fuel design. The guide thimble and instrumentation tube wall thickness is increased to improve stiffness and address incomplete rod insertion (IRI) considerations. The major outer diameter (above the dashpot) is increased to 0.482 in. from 0.474 in. for the RFA design, relative to the VANTAGE 5H, and the minor OD is increased to 0.439 in. from 0.430 in. There is no difference in major or minor (dashpot) inner diameters.

The new thimble dashpot OD (0.439 in.) requires new bottom/protective bottom grid insert tubing. The insert tube ID was increased to interface with the larger thimble tube and the guide thimble end plug was modified to a slip fit interface with the thimble tube. This results in a minimal diameter increase locally at the weld and additional margin for fit up in the insert assembly. Since the thimble tube major OD is the same as the 17x17 STD product, the RTN insert interface with the thimble tube is acceptable without a design change to the insert or lock tube. Additionally, the instrument tube socket counter bore in the debris filter bottom nozzle (DFBN) required modification to accommodate the larger instrument tube.

### Mechanical Performance

Design changes associated with the addition of the three IFM grids do not significantly influence the RFA fuel assembly structural characteristics that were determined by prior mechanical testing. Therefore, the RFA fuel assembly structural behavior and projected performance remain consistent with the VANTAGE 5H fuel assembly design.

## Core Components

The core components for BVPS are designed to be compatible with the RFA and VANTAGE 5H fuel assembly designs.

**Table 6.0-1**
**17x17 Robust Fuel Assembly and 17x17 VANTAGE 5H**
**Fuel Assembly Design**

| Design Feature | Westinghouse 17x17 VANTAGE 5H (w/o IFMs) | Westinghouse 17x17 RFA and RFA-2(1) (w/ IFMs) |
|---|---|---|
| **FUEL ASSEMBLY** | | |
| Rod Array in Assembly | 17x17 | 17x17 |
| Rods per Assembly | 264 | 264 |
| Assembly Pitch, in. | 8.466 | 8.466 |
| Overall Assembly Envelope, in. | 8.426 | 8.426 |
| Overall Assembly Height, in. | 159.775 | 159.975 |
| Fuel Assembly Weight, lb. (6" Annular Blankets) | ~1436 | ~1456 |
| Fuel Assembly Weight (Solid Blankets) | ~1469 | ~1478 |
| **BOTTOM/PROTECTIVE GRID** | | |
| Insert Tubing, OD x ID, in. | 0.4835 x 0.4455 | 0.4840 x 0.4500 |
| **BOTTOM NOZZLE** | | |
| Instrument Counter Bore Diameter, in. | 0.477 | 0.484 |
| **MID GRID** | | |
| Mid Grid Material | ZIRLO™ | ZIRLO™ |
| Mid Grid Envelope, in. | 8.418 | 8.418 |
| Vane Pattern(1) | [ ]a,c | [ ]a,c |
| Vane Length (Unbent), in.(1) | [ ]a,c | [ ]a,c |
| Spring Window, in. | [ ]a,c | [ ]a,c |
| Dimple Slot, in.(1) | [ ]a,c | [ ]a,c |
| Spring/Dimple Forms(1) | Standard Design | [ ]a,c |
| Intersect Slot Length, in. | [ ]a,c | [ ]a,c |
| Inner Strap Height, in. | 1.500 | 1.500 |
| Inner Strap Thickness, in. | 0.018 | 0.018 |
| Outer Strap Design | Standard Design | Anti-Snag |
| Outer Strap Height, in. | 1.875 | 1.878 |
| Outer Strap Thickness, in. | 0.026 | 0.026 |
| Sleeve Diameters, OD x ID, in. | 0.514 x 0.480 | 0.528 x 0.494 |
| **IFM GRID** | | |
| IFM Grid Material | N/A | ZIRLO™ |
| Envelope, in. | N/A | 8.386 |
| Vane Pattern | N/A | [ ]a,c |
| Vane Length (Unbent), in. | N/A | [ ]a,c |
| Dimple Slot, in. | N/A | [ ]a,c |
| Dimple Forms | N/A | [ ]a,c |
| Inner Strap Height, in. | N/A | 0.475 |
| Inner Strap Thickness, in. | N/A | 0.018 |
| Outer Strap Design | N/A | Anti-Snag |
| Outer Strap Height, in. | N/A | 1.363 |
| Outer Strap Thickness, in. | N/A | 0.026 |
| Sleeve Diameters, OD x ID, in. | N/A | 0.528 x 0.494 |

Note
(1) RFA and RFA-2 fuel assemblies have mid grids that differ slightly. The RFA and RFA-2 mid grid data in the table applies to both fuel assemblies, but RFA-2 mid grids have slight differences in the spring slots, width and contact face with a localized increase in dimple width at the contact face.

## 6.3 FUEL ROD DESIGN AND PERFORMANCE

### 6.3.1 Introduction

The purpose of this evaluation was to review the fuel rod design criteria to determine the acceptability of operating the BVPS fuel at the EPU core power level of 2900 MWt.

### 6.3.2 Input Parameters and Assumptions

The parameters used in the fuel rod design criteria evaluation for the EPU Project are summarized in Table 6.3-1.

### 6.3.3 Description of Analyses and Evaluations

An evaluation was performed under the EPU Project of the impact of NSSS performance parameters in Table 6.3-1 on the ability to satisfy fuel rod design criteria for BVPS.

The analyses modeled Robust Fuel Assemblies (RFA). Up to three cycles at EPU conditions were analyzed with representative rod power histories and axial power shapes generated by the NRC-approved Westinghouse advanced nodal code (ANC).

The following sections summarize the impact of the EPU core power on key fuel rod design criteria relative to their corresponding acceptance limits, and provide an assessment of the resulting impact on meeting fuel rod design criteria. The key criteria considered include rod internal pressure, clad corrosion, and clad stress and strain. Other fuel rod design criteria are not considered to be significantly impacted by the EPU core power.

#### 6.3.3.1 Rod Internal Pressure

Design Basis – The fuel system (i.e., fuel assemblies) will not be damaged due to excessive fuel rod internal pressure.

Acceptance Limit – The internal pressure of the lead rod in the reactor will be limited to a value below that which could cause the diametral gap to increase due to outward clad creep during steady state operation or for extensive DNB propagation to occur.

Design Evaluation – Margin to the rod internal pressure limit is impacted by changes in the core power rating because higher power levels result in higher fuel operating temperatures and the resulting increase in fission gas release rates. The NRC-approved Westinghouse PAD 3.4 fuel performance models, Reference 1, were used to evaluate rod internal pressure as a function of burnup. The results of this evaluation showed that rod internal pressure limits regarding gap reopening and DNB propagation can be satisfied under the assumed core duty (corresponding to a $F_{\Delta H}$ of 1.62) at the EPU conditions.

### 6.3.3.2 Clad Corrosion

Design Basis – The fuel system will not be damaged due to excessive fuel clad oxidation. The fuel system will be operated to prevent significant degradation of mechanical properties of the clad at low temperatures, as a result of hydrogen embrittlement caused by the formation of zirconium hydride platelets.

Acceptance Limit – The calculated clad temperature (metal oxide interface temperature) will be less than [ ]$^{a,c}$ °F for ZIRLO™ clad fuel ([ ]$^{a,c}$ °F for Zircaloy-4 clad fuel) during steady state operation. For Condition II events, the calculated clad temperature will not exceed [ ]$^{a,c}$ °F for ZIRLO™ clad fuel ([ ]$^{a,c}$ °F for Zircaloy-4 clad fuel). The hydrogen pickup level in the clad will be less than or equal to [ ]$^{a,c}$ ppm at the end of fuel operation.

Design Evaluation – The EPU conditions result in increased operating temperatures for the clad due to the increased rod average power rating. Since the corrosion process is a strong function of clad temperature, the EPU will impact these criteria. Using NRC-approved models, Reference 1, the impact of the EPU core power on corrosion and hydrogen pickup has been analyzed at the EPU conditions. The results of the corrosion analysis demonstrate that the corrosion limits are satisfied with the current licensed methodology.

ZIRLO™ clad fuel was analyzed and was found to meet the acceptance limits. The most likely Zircaloy-4 clad fuel (VANTAGE 5H) to be reinserted into EPU cores was similarly analyzed and was found to meet the acceptance limits. However, Zircaloy-4 clad fuel is more limiting and is therefore dependent on the specific loading pattern utilized. Cycle-specific analyses using the standard reload process will be used to demonstrate the acceptability of reinserted Zircaloy-4 clad fuel in EPU core cycles.

### 6.3.3.3 Clad Stress and Strain

Design Basis – The fuel system will not be damaged due to excessive fuel clad stress and strain.

Acceptance Limit – The volume average effective stress calculated with the Von Mises equation considering interference due to uniform cylindrical pellet-clad contact, caused by pellet thermal expansion, pellet swelling and uniform clad creep, and pressure differences, is less than the 0.2% offset yield stress with due consideration to temperature and irradiation effects under Condition II events. The acceptance limit for fuel rod clad strain during Condition II events is that the total tensile strain change due to uniform cylindrical pellet thermal expansion during a transient is less than 1% from the pre-transient value.

Design Evaluation – The Westinghouse PAD 3.4 fuel performance models, Reference 1, are used to evaluate clad stress and strain limits. The local power duty during Condition II events is a key factor in evaluating margin to clad stress and strain limits. The fuel duty at the EPU conditions is more limiting, resulting in some reduction in margin to the clad stress and strain limits. The results show that the EPU core will not impact the fuel's capability to meet clad stress and strain limits for the EPU conditions.

### 6.3.4 Acceptance Criteria and Results

The acceptance criteria and results for the fuel rod design and performance analyses and evaluations are included in Section 6.3.3.

### 6.3.5 Conclusions

The fuel rod design criteria most impacted by a change in core power rating have been reviewed with respect to the available margin to support the EPU. Although some design criteria are impacted, as stated above, the EPU conditions listed in Table 6.3-1 are supported. Finally, as in the past, cycle-specific fuel performance analysis will continue to be performed for each fuel region to confirm all fuel rod design criteria are satisfied for the operating conditions specified for each cycle of operation. These evaluations support the Reload Safety Evaluation (RSE) which is performed for each cycle of operation.

The results and conclusions of the fuel rod design and performance analyses and evaluations performed for the core power of 2900 MWt bound and support operation at the current core power of 2689 MWt, thus supporting the staged implementation of EPU at BVPS-1 and BVPS-2.

### 6.3.6 References


1. Weiner, R. A., et al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-10851-P-A (Proprietary) and WCAP-11873-A (Non-Proprietary), August 1988.

| Table 6.3-1<br>Summary of EPU Parameters<br>Analyzed in Fuel Rod Design Evaluation[1] | | |
|---|---|---|
| **Parameter** | **Current Condition** | **EPU Condition** |
| Core Power (MWt) | 2689 | 2900 |
| Core Inlet Temperature (°F) | 542.2[2] | 543.7[2] |
| Mass Flow Rate (x $10^6$, lb/hr-ft$^2$) | 2.27[2] | 2.27[2] |
| System Pressure (psia) | 2250 | 2250 |
| Cycle Lengths (MWD/MTU) | 17,000 | 20,500 |
| $F_{\Delta H}$ Limit | 1.62 | 1.62 |
| Fuel Design Considered | RFA<br>7.66 inch plenum<br>ZIRLO™ cladding<br>1.5 x IFBA (100 psia backfill) | RFA<br>7.66 inch plenum<br>ZIRLO™ cladding<br>1.5 x IFBA (100 psia backfill) |

Notes:

(1) The reinsertion of burned V5H fuel assemblies at EPU conditions would be evaluated on a cycle specific basis as part of the normal reload process.

(2) Based on minimum measured flow rate of 266,800 gpm.