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May 10, 2016

GO2-16-046

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Subject: **COLUMBIA GENERATING STATION, DOCKET NO. 50-397 LICENSE AMENDMENT REQUEST TO MODIFY TECHNICAL SPECIFICATION SURVIELLANCE REQUIREMENT (SR) 3.4.3.1 AND SR 3.4.4.1 SAFETY/RELIEF VALVES (SRVs) SETPOINT LOWER TOLERANCE**

Dear Sir or Madam:

In accordance with the provisions of Title 10 of the Code of Federal Regulation (10 CFR) Part 50.90, "Application for amendment of license, construction permit, or early site permit," Energy Northwest requests an amendment to the Technical Specifications (TS) for Columbia Generating Station (Columbia). The proposed amendment would expand the safety function lift setpoint tolerances for the Safety/Relief Valves (SRVs) that are listed in Surveillance Requirements (SRs) 3.4.3.1 and 3.4.4.1 of the TS. This change would be limited to the lower tolerances and would not affect the upper limits. The tolerance band for these valves would be changed from ±3% to +3%, -5% of the setpoint.

The as-left tolerance band for the safety function lift setpoints will continue to be ±3%, but the as-found tolerance band will change to +3%, -5%. If a valve is tested and the lift setpoint is found outside the tolerances, the valve failure will be evaluated in the Columbia Corrective Action Program. The as-found tolerance band is used for determining operability and to increase sample size for Inservice Testing (IST).

The proposed change has been evaluated in accordance with 10 CFR 50.91(a)(1) using criteria in 10 CFR 50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in Attachment 1 of this submittal.

The proposed TS markup pages are included as Attachment 2 to this submittal. Markups of the proposed TS Bases are included for information only as Attachment 3 of this submittal. Clean pages of the proposed TS changes are included as Attachment 4 of this submittal.

Energy Northwest requests approval of the proposed license by April 30, 2017 to support implementation during the next refueling outage. Once approved, the amendment will be implemented within 60 days. This implementation period will provide adequate time for the affected station documents to be revised using the appropriate change control mechanisms.

This application contains no new commitments.

In accordance with 10 CFR 50.91, Energy Northwest is notifying the State of Washington of this amendment request by transmitting a copy of this letter and attachments to the designated State Official.

If there are any questions or if additional information is needed, please contact Ms. L. L. Williams, Licensing Supervisor, at 509-377-8148.

I declare under penalty of perjury that the foregoing is true and correct. Executed this 10th day of May, 2016.

Respectfully,

A. L. Javorik
Vice President, Engineering

Attachments: As stated

cc: NRC Region IV Administrator
NRC NRR Project Manager
NRC Sr. Resident Inspector - 988C
CD Sonoda - BPA1399 (email)
WA Horin - Winston & Strawn
RR Cowley - WDOH (email)
EFSECutc.wa.gov- EFSEC (email)

Evaluation of Proposed Technical Specification (TS) Change

1. SUMMARY OF DESCRIPTION

The proposed amendment will revise the Columbia Generating Station (Columbia) Technical Specification (TS) Surveillance Requirements (SRs) 3.4.3.1 and 3.4.4.1 to change the safety function lift setpoint lower tolerance limit from -3% to -5% for the Safety/Relief Valves (SRVs). This change would be limited to the lower tolerance and does not affect the upper limit. This change only applies to the as-found tolerance and not to the as-left tolerance band which will remain at ±3**%** of the safety function lift setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for testing. There will be no change to the actual setpoints of the valves installed in the plant. The as-found safety function lift setpoint lower tolerance for the SRVs will be revised from -3% to -5%.

The proposed amendment will only be applicable to the safety mode of operation which is defined as the mode which is dependent only on system pressure, i.e. the spring mode of operation.

2. DETAILED DESCRIPTION

The proposed amendment would change the safety function lift setpoint lower tolerance for the SRVs that are listed in TS:

- 3.4.3, Safety/Relief Valves (SRVs) ≥ 25% Rated Thermal Power (RTP), SR 3.4.3.1 and

- 3.4.4, Safety/Relief Valves (SRVs) < 25% RTP, SR 3.4.4.1.

The tolerance band for SRVs would be changed from ±3**%** to +3, -5% of the safety lift function setpoint. This change only applies to the as-found tolerance band and not to the as-left tolerance band which will remain at ±3% of the safety function lift setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for testing. There will be no change to the valves as installed in the plant. The following are the current and proposed setpoint tolerance values which apply to both SR 3.4.3.1 and SR 3.4.4.1:

Number of SRVs	Current Setpoint (psig)	Proposed Setpoint (psig)
2	1165 ±34.9	1165 +34.9 -58.2
4	1175 ±35.2	1175 +35.2 -58.7
4	1185 ±35.5	1185 +35.5 -59.2
4	1195 ±35.8	1195 +35.8 -59.7

Number of SRVs	Current Setpoint (psig)	Proposed Setpoint (psig)
4	1205 ±36.1	1205 +36.1 -60.2

The proposed change reduces unnecessarily restrictive Surveillance Requirements, which result in additional SRV testing to meet American Society of Mechanical Engineers (ASME) OM Code requirements (Reference 12). The proposed change will establish an “Owner-established” set-pressure ± tolerance limit. The ASME OM Code, Subsection ISTC-5240 requires testing of safety and relief devices in accordance with Mandatory Appendix I. Mandatory Appendix I, paragraph I-1320(c)(1) requires two additional valves to be tested for each valve that exceeds its as-found set-pressure by the greater of either the ± tolerance limit of the Owner-established set-pressure acceptance criteria of I-1310(e) or ±3% of the valve nameplate set-pressure.

The revision will only be applicable to the safety mode of operation which is defined as the mode which is dependent only on system pressure, i.e. the spring mode of operation. The relief mode of SRV operation is dependent upon external power sources and will remain unchanged. The setpoints for the Automatic Depressurization System (ADS) mode of operation will remain unchanged.

3. BACKGROUND

At Columbia Generating Station (Columbia), the reactor pressure vessel (RPV) and appurtenances and the reactor coolant pressure boundary (RCPB) piping, pumps and valves, were designed, fabricated, and tested with the applicable edition of the ASME Boiler and Pressure Vessel Code Section III, including the Addenda that were mandated by Title 10 of the Code of Federal Regulations (10 CFR) 50.55a at the order date for the applicable components. Each SRV is a Crosby 6 R 10, Style HB-65-BP, originally built to ASME Section III, 1971 Edition with no Addenda.

The ASME Boiler and Pressure Vessel Code, Section III (Reference 1) requires the RPV be protected from overpressure during upset conditions by self-actuated safety valves. As part of the nuclear pressure relief system, the size and number of SRVs are selected such that peak pressure in the nuclear system will not exceed the ASME Code limits for the RCPB.

3.1 SRV Safety Mode

The SRVs can actuate by either of two modes: the safety mode or the relief mode. In the safety mode, or spring mode of operation, the direct action of the steam pressure in the main steam lines will act against a spring loaded disk that will open when the valve inlet pressure exceeds the spring force.

The Columbia SRV safety mode setpoints are established to ensure the ASME Boiler and Pressure Vessel Code, Section III, limit on peak reactor pressure is satisfied. The ASME Boiler and Pressure Vessel Code, Section III, requires the lowest safety valve be set at or below the reactor pressure vessel design pressure of 1250 psig and the

highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for over pressurization conditions. The transient evaluations in the Columbia Final Safety Analysis Report (FSAR) Section 5.2.2.2 (Reference 2) as well as the analyses in GE-NE-187-24-0992, " Washington Public Power Supply System Nuclear Project 2 SRV Setpoint Tolerance and Out-of-Service Analysis," Revision 2, (Reference 3) and NEDC-32115P, " Washington Public Power Supply System Nuclear Project 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, (Reference 4) involving the safety mode are based on these setpoints. These analyses also include additional uncertainties of ± 3% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.

The Columbia overpressure protection report required by ASME Boiler and Pressure Vessel Code Section III, Article NB-7000, Protection against Overpressure, was completed using the ASME Code Section III 1971 Edition up to and including Summer 1971 Addenda.

3.2 SRV Relief Mode

The second mode the SRVs can actuate is the relief mode. In the relief mode or power actuated mode of operation, a pneumatic piston/cylinder and mechanical linkage assembly are used to open the valve by overcoming the spring force, even with the valve inlet pressure equal to 0 psig. The pneumatic operator is arranged so that its malfunction will not prevent the valve disk from lifting if steam inlet pressure reaches the spring lift set pressures. In the relief mode, valves may be opened manually or automatically at the selected preset pressure. Seven of the SRVs that provide the safety and relief function are part of the ADS. The SRV relief mode will actuate prior to the safety mode. Although no credit is taken under ASME Boiler and Pressure Vessel Code Section III for overpressure protection by the SRVs in the relief mode, the relief mode will limit peak reactor pressure in the majority of overpressure transients.

3.3 ADS Mode

The ADS functions to reduce the reactor pressure so that flow from low pressure core injection (LPCI) and the low pressure core spray system (LPCS) enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the SRVs to relieve the high pressure steam to the suppression pool.

3.4 SRV Test Method

The Columbia SRVs have remote set-pressure verification devices (SPVDs) permanently installed to allow set-pressure testing at low power operation, typically during shutdown for refueling outage and on startup if necessary. Each SPVD incorporates a nitrogen powered, metal bellows assembly that adds a quantified lifting force on the valve stem until the SRV's lift pressure is reached. During normal power operation, these SPVDs remain de-energized and do not interfere with normal safety or relief valve functions. Removal and replacement of the SRVs is normally performed only for valve maintenance and not for the purpose of as-found set-pressure determination. Some SRVs are removed and replaced for maintenance purposes such as seat

leakage, or refurbishment nominally each refueling outage. The valves which are required to be as-found set-pressure tested, as part of the ASME OM Code required periodic testing, do not necessarily correspond to those required to be replaced for maintenance.

The SRVs are as-found tested per the Inservice Testing Program (IST). A minimum of 20% of the SRVs (4 valves) are required to be tested in any 24-month interval. Each installed SRV is to be as-found tested at least once every 5 years with the test interval defined from test to test date. To meet the 5-year interval requirement, at least 9 SRVs are tested each refueling outage. When as-found test requirements have been satisfied for a given 24-month or 5-year test interval, additional valves removed for maintenance do not require as-found set-pressure testing prior to disassembly for maintenance.

4. TECHNICAL EVALUATION

4.1 History of Technical Specification Requirements

The current SRV safety setpoint tolerance band comes from the Columbia LAR to increase the licensed power level from 3323 MWt to 3486 MWt and change the SRV setpoint tolerance band from +1%, -3% to ±3%. The tolerance band change was supported by an analysis performed by GE Nuclear Energy (Reference 3). It also included a 15 psi Group 1 SRV setpoint increase for the power uprate (Reference 5). This LAR was approved in 1995 by the NRC as Amendment 137 (Reference 6).

4.2 Operating Experience and Manufacturer Recommendations

As discussed earlier, the ASME OM Code, Mandatory Appendix I, requires two additional valves to be tested for each valve that exceeds its as-found set-pressure by the greater of either the ± tolerance limit of the Owner-established set-pressure acceptance criteria of I-1310(e) or ±3% of the valve nameplate set-pressure. The proposed change will establish an “Owner-established” set-pressure ± tolerance band.

The tendency to lift low is a known operating characteristic of the Crosby valve. The Crosby SRV operating and setpoint test history at Columbia is consistent with industry operating experience. The valves are often slightly low on their as-found setpoint. Setpoint tests which are outside the acceptance criteria are evaluated in the Columbia Corrective Action Program. Subsequent setpoint tests performed in accordance with ASME OM Code I-4110(i) are usually within normal setpoint tolerance range.

The Crosby Valve and Gage Company Procedure I-11069, “Instruction Manual for Crosby Style 6xRx10 HB-65-BP Safety Relief Valve for Main Steam Service,” Revision 1 (Reference 14), Section 3, step 3.1.1.4 states:

> “Setpoint repeatability for on line service should be within a tolerance of +1%, -3% of nameplate pressure. However, the low limit setpoint tolerance may be extended to 1067 psig if all other valve functional requirements are met.”

The proposed SRV setpoint lower tolerance change from -3% to -5% will lower the minimum TS SR 3.4.3.1 and TS SR 3.4.1.1 SRV as-found acceptance criteria to 1106.8 psig (1165-58.2). This new minimum as-found acceptance criterion is greater than the manufacturer's recommended low setpoint pressure of 1067 psig.

4.3 Surveillance Test History

Table 1 shows the surveillance testing results for the SRVs from 2005 to 2015. The average drift of the 97 SRV tests shown in Table 1 is -0.88% with a median of -0.43%. There were 16 valves that failed below the -3% tolerance. There was only one valve which had a greater than -5% deviation from setpoint.

The Columbia SRV setpoint test history for all valves shows that a failure of the valve to lift low during one outage is not adequate to predict that the valve will lift low during the next test interval.

Table 1

SRV Equipment No.	Year/Outage	Test Type	SRV S/N	Set Pressure	Tolerance	As-found	Deviation
1A	2005, R17	C	47	1175	+/- 3%	1165	-0.85%
1B	2005, R17	C	140	1165	+/- 3%	1132	-2.83%
1C	2005, R17	C	45	1165	+/- 3%	1102	-5.41%
1D	2005, R17	CA	122	1175	+/- 3%	1140	-2.98%
2A	2005, R17	CA	51	1185	+/- 3%	1195	0.84%
2B	2005, R17	C	49	1175	+/- 3%	1137	-3.23%
2C	2005, R17	C	48	1175	+/- 3%	1162	-1.11%
2D	2005, R17	C	52	1185	+/- 3%	1198	1.10%
3A	2005, R17	CA	58	1195	+/- 3%	1180	-1.26%
3B	2005, R17	C	53	1185	+/- 3%	1148	-3.12%
3C	2005, R17	C	124	1185	+/- 3%	1158	-2.28%
3D	2005, R17	CA	57	1195	+/- 3%	1193	-0.17%
4A	2005, R17	C	59	1205	+/- 3%	1180	-2.07%
4B	2005, R17	C	137	1195	+/- 3%	1210	1.26%
4C	2005, R17	C	55	1195	+/- 3%	1218	1.92%
4D	2005, R17	C	61	1205	+/- 3%	1195	-0.83%
5B	2005, R17	CA	60	1205	+/- 3%	1195	-0.83%
5C	2005, R17	C	136	1205	+/- 3%	1195	-0.83%
1A	2007, R18	C	47	1175	+/- 3%	1175	0.00%
1B	2007, R18	C	140	1165	+/- 3%	1137	-2.40%
2A	2007, R18	C	51	1185	+/- 3%	1196	0.93%
2B	2007, R18	C	49	1175	+/- 3%	1176	0.09%
3A	2007, R18	C	58	1195	+/- 3%	1205	0.84%
3B	2007, R18	C	53	1185	+/- 3%	1174	-0.93%
4A	2007, R18	C	59	1205	+/- 3%	1200	-0.41%
4B	2007, R18	C	137	1195	+/- 3%	1215	1.67%
5B	2007, R18	C	60	1205	+/- 3%	1196	-0.75%
1A	2009, R19	CA	50	1175	+/- 3%	1174	-0.09%
1C	2009, R19	C	45	1165	+/- 3%	1166	0.09%

Table 1							
SRV Equipment No.	**Year/Outage**	**Test Type**	**SRV S/N**	**Set Pressure**	**Tolerance**	**As-found**	**Deviation**
1D	2009, R19	C	122	1175	+/- 3%	1119	-4.77%
2A	2009, R19	CA	51	1185	+/- 3%	1189	0.34%
2C	2009, R19	C	48	1175	+/- 3%	1129	-3.91%
2D	2009, R19	C	54	1185	+/- 3%	1160	-2.11%
3A	2009, R19	CA	58	1195	+/- 3%	1168	-2.26%
3B	2009, R19	CA	138	1185	+/- 3%	1185	0.00%
3C	2009, R19	C	124	1185	+/- 3%	1190	0.42%
3D	2009, R19	C	57	1195	+/- 3%	1200	0.42%
4A	2009, R19	CA	135	1205	+/- 3%	1211	0.50%
4B	2009, R19	C	126	1195	+/- 3%	1190	-0.42%
4C	2009, R19	C	56	1195	+/- 3%	1214	1.59%
4D	2009, R19	C	61	1205	+/- 3%	1161	-3.65%
5B	2009, R19	CA	60	1205	+/- 3%	1214	0.75%
5C	2009, R19	C	136	1205	+/- 3%	1186	-1.58%
1A	2011, R20	C	50	1175	+/- 3%	1146	-2.47%
1B	2011, R20	C	139	1165	+/- 3%	1172	0.60%
1C	2011, R20	CA	46	1165	+/- 3%	1187	1.89%
1D	2011, R20	CA	49	1175	+/- 3%	1191	1.36%
2A	2011, R20	C	51	1185	+/- 3%	1185	0.00%
2B	2011, R20	C	134	1175	+/- 3%	1161	-1.19%
2C	2011, R20	CA	47	1175	+/- 3%	1174	-0.09%
2D	2011, R20	CA	54	1185	+/- 3%	1132	-4.47%
3A	2011, R20	C	58	1195	+/- 3%	1158	-3.10%
3B	2011, R20	C	138	1185	+/- 3%	1154	-2.62%
3C	2011, R20	CA	53	1185	+/- 3%	1156	-2.45%
3D	2011, R20	CA	137	1195	+/- 3%	1173	-1.84%
4A	2011, R20	C	135	1205	+/- 3%	1186	-1.58%
4B	2011, R20	C	126	1195	+/- 3%	1190	-0.42%
4C	2011, R20	CA	56	1195	+/- 3%	1201	0.50%
4D	2011, R20	CA	62	1205	+/- 3%	1148	-4.73%
5B	2011, R20	C	60	1205	+/- 3%	1208	0.25%
5C	2011, R20	CA	59	1205	+/- 3%	1168	-3.07%
1A	2013, R21	CA	48	1175	+/- 3%	1123	-4.43%
1B	2013, R21	CA	45	1165	+/- 3%	1143	-1.89%
1C	2013, R21	C	46	1165	+/- 3%	1136	-2.49%
1D	2013, R21	C	49	1175	+/- 3%	1193	1.53%
2A	2013, R21	CA	122	1185	+/- 3%	1188	0.25%
2B	2013, R21	CA	134	1175	+/- 3%	1166	-0.77%
2C	2013, R21	C	47	1175	+/- 3%	1166	-0.77%
2D	2013, R21	C	52	1185	+/- 3%	1152	-2.78%
3A	2013, R21	CA	57	1195	+/- 3%	1190	-0.42%
3B	2013, R21	CA	138	1185	+/- 3%	1180	-0.42%
3C	2013, R21	C	53	1185	+/- 3%	1183	-0.17%
3D	2013, R21	C	137	1195	+/- 3%	1210	1.26%
4A	2013, R21	CA	135	1205	+/- 3%	1204	-0.08%
4B	2013, R21	CA	126	1195	+/- 3%	1196	0.08%

Table 1							
SRV Equipment No.	**Year/Outage**	**Test Type**	**SRV S/N**	**Set Pressure**	**Tolerance**	**As-found**	**Deviation**
4C	2013, R21	C	55	1195	+/- 3%	1257	5.19%
4D	2013, R21	C	62	1205	+/- 3%	1162	-3.57%
5B	2013, R21	CA	61	1205	+/- 3%	1208	0.25%
5C	2013, R21	C	59	1205	+/- 3%	1173	-2.66%
1A	2015, R22	CA	48	1175	+/- 3%	1122	-4.51%
1B	2015, R22	C	45	1165	+/- 3%	1138	-2.32%
1C	2015, R22	CA	139	1165	+/- 3%	1146	-1.63%
1D	2015, R22	CA	49	1175	+/- 3%	1209	2.89%
2A	2015, R22	CA	122	1185	+/- 3%	1185	0.00%
2B	2015, R22	C	134	1175	+/- 3%	1170	-0.43%
2C	2015, R22	CA	50	1175	+/- 3%	1134	-3.49%
2D	2015, R22	C	52	1185	+/- 3%	1133	-4.39%
3A	2015, R22	C	57	1195	+/- 3%	1157	-3.18%
3B	2015, R22	C	138	1185	+/- 3%	1180	-0.42%
3C	2015, R22	CA	54	1185	+/- 3%	1180	-0.42%
3D	2015, R22	C	137	1195	+/- 3%	1208	1.09%
4A	2015, R22	C	135	1205	+/- 3%	1200	-0.41%
4B	2015, R22	C	126	1195	+/- 3%	1186	-0.75%
4C	2015, R22	CA	56	1195	+/- 3%	1277	6.86%
4D	2015, R22	CA	62	1205	+/- 3%	1173	-2.66%
5B	2015, R22	C	61	1205	+/- 3%	1226	1.74%
5C	2015, R22	CA	136	1205	+/- 3%	1214	0.75%

Table Key:
S/N= Serial Number
Test Type:
C- Test for ASME Code and TS Compliance
CA- Additional SRVs tested due to ASME Code Test failure

4.4 Operating Margin

The purpose of the lower setpoint tolerance is to ensure sufficient margin exists between the normal operating pressure of the system and the point at which the SRVs actuate in the safety mode. The normal operating pressure of the RPV at power is 1020 psig. Table 2 provides a comparison of the SRV safety setpoints with the relief setpoints and provides the simmer margin between the proposed -5% safety setpoint tolerance and the normal operating pressure. The table shows there is adequate margin between the normal operating pressure and the -5% safety setpoint tolerance. The lowest margin occurs with the lowest SRV setpoint valves with a simmer margin of 86.75 psig.

The table also shows a comparison with the relief mode setpoints along with their established tolerance band of ± 5 psig. The table shows there is the potential for slight overlap with the relief setpoints for the proposed safety setpoint lower tolerance of -5%. However, the as-left tolerance of SRVs remains at ±3%. SRVs which are removed for maintenance are returned to a tolerance of ±1% prior to being installed for service, thereby returning the margin to original levels. Therefore, the margin is considered adequate and will not impact normal plant operation.

Additionally, for the Columbia analyses discussed in section 4.5, there is no specific margin required between event peak pressure and opening safety setpoints for the SRVs or a restriction in the overlap between the range of acceptable SRV safety setpoints for the SRVs and the relief mode opening setpoints.

Table 2

SRV Equip No.	TS Safety Setpoint	TS Safety -3%	TS Safety -5%	Pressure Switch Equip. No.	Relief Setpoint	Relief -5 psig	Relief -5 psig vs Safety -3%	Relief -5 psig vs Safety -5%	Simmer Margin for -5%
1A	1175	1139.8	1116.25	39J	1120	1115	24.8	1.25	96.25
1B	1165	1130.1	1106.75	39E	1111	1106	24.1	0.75	86.75
1C	1165	1130.1	1106.75	39L	1110	1105	25.1	1.75	86.75
1D	1175	1139.8	1116.25	39K	1120	1115	24.8	1.25	96.25
2A	1185	1149.5	1125.75	39A	1131	1126	23.5	-0.25	105.75
2B	1175	1139.8	1116.25	39F	1121	1116	23.8	0.25	96.25
2C	1175	1139.8	1116.25	39D	1121	1116	23.8	0.25	96.25
2D	1185	1149.5	1125.75	39C	1131	1126	23.5	-0.25	105.75
3A	1195	1159.2	1135.25	39B	1141	1136	23.2	-0.75	115.25
3B	1185	1149.5	1125.75	39H	1130	1125	24.5	0.75	105.75
3C	1185	1149.5	1125.75	39G	1131	1126	23.5	-0.25	105.75
3D	1195	1159.2	1135.25	39V	1141	1136	23.2	-0.75	115.25
4A	1205	1168.9	1144.75	39S	1151	1146	22.9	-1.25	124.75
4B	1195	1159.2	1135.25	39R	1141	1136	23.2	-0.75	115.25
4C	1195	1159.2	1135.25	39M	1141	1136	23.2	-0.75	115.25
4D	1205	1168.9	1144.75	39P	1151	1146	22.9	-1.25	124.75
5B	1205	1168.9	1144.75	39U	1151	1146	22.9	-1.25	124.75
5C	1205	1168.9	1144.75	39N	1151	1146	22.9	-1.25	124.75

4.5 Evaluation

The proposed safety setpoint lower tolerance change from -3% to -5% was qualitatively evaluated for Columbia by GE Hitachi Nuclear Energy (GEH) in Reference 7 as a supplement to GE-NE-187-24-0992, Revision 2. The results of this evaluation are presented below.

4.5.1 Vessel overpressure protection, anticipated operational occurrence (AOO) thermal limits and anticipate transient without scram (ATWS):

The effects of the proposed change on ATWS and other pressurization events are evaluated below.

The AOO thermal limits evaluation examined the Columbia Cycle 23 analysis. For the events that establish the operating limit minimum critical power ratio (OLMCPR) and the

off-rated limits, later opening of the SRVs provides the most conservative results. The pressure feedback on the core is larger with later opening times such that the power feedback is worse, and as a result, the OLMCPR would be worse if the time of the minimum critical power ratio (MCPR) was after the SRV opening time. Since the proposed change for the SRV safety mode setpoint lower tolerance from -3% to -5% would allow the SRVs to open sooner, this results in a benefit or no change to the OLMCPR and off-rated limits.

For the events that establish the peak vessel pressures for the ASME overpressure protection, later opening of the SRVs provides the most conservative results. The objective of these events is to maximize vessel dome and thus vessel bottom pressures. Since the proposed change to decrease the SRV safety mode setpoint lower tolerance from -3% to -5% would allow opening of the SRVs earlier, this will result in a benefit to the ASME overpressure protection analyses.

For the ATWS analyses, nominal values for the relief mode of the SRVs are used. Since the proposed change is only for the SRV safety mode setpoint tolerance and does not propose any change to the SRV relief mode setpoints, the impact on the ATWS analyses is negligible.

Additionally, an ATWS analysis result, peak RPV pressure for the period of Standby Liquid Control (SLC) injection, is used in the SLC pump discharge relief valve setting determination. Since the ATWS evaluation above determined the ATWS analysis is unaffected by the SRV safety mode setpoint tolerance from -3% to -5% change, the peak RPV pressure during ATWS for the period of SLC injection is not changed. Since the proposed change to decrease the SRV safety mode setpoint lower tolerance does not change the peak RPV pressure for an ATWS event during SLC injection, the basis for the SLC pump discharge relief valve setting is not affected by this change.

4.5.2 <u>High Pressure Systems</u>:

The impact of the setpoint tolerance change on the performance characteristics of the High Pressure Core Spray (HPCS), Reactor Core Isolation Cooling (RCIC), and SLC systems is evaluated below.

The HPCS system is a safety system and part of the emergency core cooling system (ECCS) for Columbia. HPCS provides high pressure water injection into the RPV in the event of a loss of coolant accident (LOCA). Additionally, HPCS is a backup to the RCIC system to provide makeup water in the event of a loss of feedwater (LOFW) flow transient.

For the LOCA event, credit is taken for the safety mode of the SRV, and the analysis uses the opening pressures of the SRV safety mode setpoint upper tolerance of +3%. Since the proposed change is a decrease in SRV safety mode setpoint lower tolerance from -3% to -5%, there is no impact on the HPCS system.

For LOFW events that do not isolate the RPV, the RPV pressure is maintained by the turbine bypass valves (TBVs) at or near the pressure regulator setpoint. With the RPV

pressures near the pressure regulator setpoint, HPCS operation is not affected by changes in the SRV performance. For LOFW events with a closure of main steam isolation valves (MSIVs), the RPV pressure is dictated by the opening setpoint of the relief mode of the SRVs. Since the proposed change is only for the SRV safety mode setpoint lower tolerance and does not propose any change to the SRV relief mode setpoints, there is no impact on the HPCS system for LOFW.

The RCIC system is designed to maintain RPV water level above the RPV Water Level–Low Low Low, Level 1 setpoint in the event of a LOFW flow transient. For LOFW events that do not isolate the RPV, the RPV pressure is maintained by the TBVs at or near the pressure regulator setpoint. With the RPV pressures near the pressure regulator setpoint, RCIC operation is not affected by changes in the SRV performance. For LOFW events with a closure of the MSIVs, the RPV pressure is dependent on the relief mode of the SRVs. Since the proposed change is only for the SRV safety mode setpoint lower tolerance and does not propose any change to the SRV relief mode setpoints, there is no impact on the RCIC system.

The SLC system is designed to shut down the reactor from power conditions to cold shutdown in the postulated situation of a failure of the control rods to insert. The SLC system is designed for injection at a maximum RPV pressure equal to the upper analytical limit for the lowest group of SRVs opening in the relief mode. Since the proposed change is only for the SRV safety mode setpoint lower tolerance and does not propose any change to the SRV relief mode setpoints, there is no impact on the SLC system.

4.5.3 Containment Response:

The proposed change to the lower SRV safety mode setpoint tolerance from -3% to -5% has no impact on the short term or long term containment analyses. This change has an impact only on the SRV containment dynamic loads from the containment components addressed in References 3, 9, and 13. Since flow is proportional to pressure, the SRV flow rates will increase linearly with the SRV opening pressure and the SRV discharge line and quencher loads will increase linearly with SRV flow rate. Since the proposed change in the SRV safety mode tolerance from -3% to -5% would allow the SRV to open at a lower pressure resulting in lower flow, and lower discharge line and quencher line loads, the current SRV discharge line and quencher loads are not impacted by the proposed change. In addition, the suppression pool boundary pressure loads and submerged structure loads are not impacted by the change in the SRV setpoints due to the lower limit tolerance change from -3% to -5% (References 9 and 13).

4.5.4 Loss of Coolant Accident (LOCA):

The Columbia ECCS-LOCA analysis credits the SRV safety mode setpoint upper tolerance and the ADS capacity. The proposed change to the SRV safety mode lower tolerance from -3% to -5% does not impact the SRV safety mode setpoint upper tolerance or the ADS capacity. Since the proposed change to the safety mode setpoint lower tolerance was not used in the ECCS-LOCA analysis, there is no impact on a LOCA event.

5. REGULATORY EVALUATION

The Columbia FSAR Chapter 3 provides detailed discussion of Columbia's compliance with the applicable regulatory requirements and guidance.

The proposed TS amendment:

- Does not result in any change in the qualifications of any component; and
- Does not result in the reclassification of any component's status in the areas of shared, safety-related, independent, redundant, and physically or electrically separated.

5.1 Applicable Regulatory Requirements

5.1.1. Applicable 10 CFR 50 Appendix A General Design Criteria (GDC)

The following GDCs for the Reactor Coolant System, which require that the system be protected from over-pressurization, were evaluated to determine if these GDC continue to be met.

GDC 15 - Reactor Coolant System Design (Criterion 15)

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

An example of the integrated protective action scheme, which provides sufficient margin to ensure that the design conditions of the RCPB are not exceeded, is the automatic initiation of the nuclear system pressure relief system on receipt of an overpressure signal. To accomplish overpressure protection, a number of pressure-operated relief valves are provided that can discharge steam from the nuclear system to the suppression pool. The nuclear system pressure relief system also provides for automatic depressurization of the nuclear system in the event of a LOCA in which the vessel is not depressurized by the accident. The depressurization of the nuclear system in this situation allows operation of the low-pressure ECCS to supply enough cooling water to adequately cool the core.

The proposed change only affects the lower SRV setpoint tolerance of the safety setpoint function of the SRVs. The lower setpoint tolerance change from -3% to -5% does not impact the design of the RCPB since the setpoint tolerance for the upper safety setpoint tolerance remains unchanged. Additionally, the relief mode of operation of the SRV and the ADS mode of operation are not affected by the proposed change. Therefore, Criterion 15 is not affected by the proposed change.

GDC 35 – Emergency Core Cooling (Criterion 35)

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The ADS functions to reduce the reactor pressure so that flow from low pressure core injection (LPCI) and the low pressure core spray system (LPCS) enters the reactor vessel in time to cool the core and prevent excessive fuel clad temperature. The ADS uses several of the SRVs to relieve the high pressure steam to the suppression pool. The ADS function of the SRVs remains unchanged by the proposed change to the lower safety setpoint tolerance for the safety function of the SRVs. Therefore, Criterion 35 is not affected by the proposed change.

5.1.2. Applicable ASME Code Requirements

The proposed change has been evaluated to determine whether application of regulations and requirements continue to be met.

The Columbia IST Program is currently implemented in accordance with ASME Operation and Maintenance (OM) Code 2004 Edition through 2006 Addenda. The SRVs are Class 1 Category C valves in accordance with the Columbia IST Program which tests the SRVs in accordance with the ASME OM Code Subsection ISTC-5240 and Mandatory Appendix I.

Columbia currently has one approved relief request RV03 which provides an alternative to ASME OM Code Mandatory Appendix I, Paragraph I-3310, “Sequence of Periodic Testing of Class 1 Main Steam Pressure Relief Valves and Auxiliary Actuating Devices.” RV03 provides an alternate to Mandatory Appendix I, paragraph I-3310, “Sequence of Periodic Testing of Class 1 Main Steam Pressure Relief Valves with Auxiliary Actuating Devices.” This relief allows the following:

1. "Valves" and "accessories" (actuators, solenoids, etc.) shall be tested separately and meet Paragraph I-1320 test frequency requirements. Since the valve and actuator test and maintenance cycles are different, the plant positions of the actuators selected, or due, for periodic testing may not match the plant positions of the MSRVs selected, or due, for as-found set-pressure testing.

[Main Steam Relief Valve] MSRV periodic set-pressure testing will normally be performed at power during shutdown for refueling outage. As-found visual

examination will be performed after set-pressure testing which is out of specified Code required sequence. If MSRV periodic set-pressure testing could not be performed at power during shutdown for refueling outage due to reactor scram it will be required to be performed during power ascension from refueling outage or by removing the valves and sending them to the vendor for as-found set-pressure testing. This will require Paragraphs I-3310(a), (d) and (e) tests to be performed during outage prior to Paragraphs I-3310(b), (c) and (i) tests.

The actuators and solenoids will be tested at the end of the outage after other maintenance is complete, and the tests will be credited as satisfying the Code periodic test requirements provided that no actuator or solenoid maintenance (other than actuator assembly re-installation on a replaced valve) is performed that would affect their as-found status prior to testing or that could affect the valve’s future set-pressure determination.

2. All MSRV position indicators will continue to be tested in accordance with existing surveillance procedures for monthly channel checks, and for channel calibration and channel functional testing at least once per 24 months during shutdowns. These tests will be credited for satisfying the requirements of Paragraph I-3310(f).

3. All auxiliary actuating device sensing elements (pressure switches) will continue to be tested and calibrated on a 24 month frequency. These tests will be credited for satisfying the requirements of paragraph I-3310(h).

Relief request RV03 was approved by the NRC on December 9, 2014 (Reference 8). The proposed change to the Owner-established set pressure ± tolerance band does not impact the approved relief request.

5.2 Precedent

There are two precedents where plants have changed the lower setpoint tolerance for the safety function of the plant’s SRVs from -3% to -5%. The first is Susquehanna Steam Electric Station, Units 1 and 2 that was granted November 17, 2011 (Reference 10). The Susquehanna Steam Electric Station, Units 1 and 2 are BWR plants with the same SRV manufacturer and model as the Columbia SRVs.

The second precedent is for River Bend Station, Unit 1 that was granted February 13, 2003 (Reference 11). River Bend Station, Unit 1 is a BWR plant with the same SRV manufacturer but larger model SRV than the Columbia SRV model. The River Bend SRV is a Crosby 8 X10. Columbia’s SRV is Crosby 6 X R X 10. Although the River Bend model is slightly different, the basic design and operation of the valve is essentially the same.

5.3 No Significant Hazards Consideration Determination

The proposed amendment will revise the Columbia TS SRs 3.4.3.1 and 3.4.4.1 to change the safety function lift setpoint lower tolerance limit for the SRVs. This change

would be limited to the lower tolerances and does not affect the upper limits. This change only applies to the as-found tolerance and not to the as-left tolerance which will remain at ±3**%** of the safety lift function setpoint. The as-found tolerances are used for determining operability and to increase sample sizes for testing. There will be no change to the actual setpoints of the valves installed in the plant. The as-found safety function lift setpoint lower tolerance for the SRVs will be revised from -3% to -5%.

Energy Northwest has evaluated whether or not a significant hazards consideration is involved with the proposed amendment as set forth in 10 CFR 50.92, “Issuance of amendment,” as discussed below.

5.3.1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No

This proposed amendment has no influence on the probability or consequences of any accident previously evaluated. The lower safety setpoint tolerance change does not affect the operation of the SRVs and it does not affect the as-left setpoint tolerance band which is unchanged at ±3% of the lift setpoint of the SRVs. The change only affects the lower tolerance for opening of the SRVs. The proposed amendment does not affect the upper tolerance for SRVs safety setpoints, which is the limit that protects from overpressurization.

The proposed amendment does not involve any physical changes to the SRVs, nor does it change the safety function of the SRVs. The proposed TS revision involves no significant changes to the operation of any systems or components in normal or accident operating conditions as discussed in the technical evaluation for this LAR. Additionally, the proposed change does not involve any significant changes to existing structures, systems, or components.

The proposed amendment does not change any other behavior or operation of the SRVs, and, therefore, has no significant impact on reactor operation. It also has no significant impact on response to any perturbation of reactor operation including transients and accidents previously analyzed in the FSAR.

Therefore, the proposed amendment does not result in a significant increase in the probability or consequences of any previously evaluated accident.

5.3.2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No

The proposed change from -3% to -5% for the SRV safety setpoint lower tolerance only affects the criteria to determine when an as-found SRV test is considered acceptable. The proposed change does not affect the criteria for the setpoint upper tolerance for the SRVs.

The proposed change from -3% to -5% for the SRV safety setpoint lower tolerance does not adversely affect the operation of any safety-related components or equipment. Since the proposed amendment does not involve any hardware changes, significant changes to the operation of any systems or components, nor change to existing structures, systems, or components, there is no possibility that a new or different kind of accident is created.

The proposed change from -3% to -5% for the SRV safety setpoint lower tolerance does not involve any physical changes to the SRVs, nor does it change the safety function of the SRVs. The proposed change does not require any physical change or alteration of any existing plant equipment. No new or different equipment is being installed. No installed equipment is being operated in a new or different manner. There is no alteration to the parameters within which the plant is normally operated. This change does not alter the manner in which equipment operation is initiated, nor will the functional demands on credited equipment be changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed. No changes are being made to the procedures relied upon to respond to off-normal events as described in the FSAR are being proposed by this change. The proposed change does not alter assumptions made in the safety analysis and licensing basis.

Therefore, the proposed amendment does not create the possibility of a new or different kind of accident from any previously evaluated.

5.3.3. Does the proposed amendment involve a significant reduction in margin of safety?

Response: No

The proposed change from -3% to -5% for the SRV safety setpoint lower tolerance only affects the criteria to determine when an as-found SRV test is considered acceptable. This change does not affect the criteria for the SRV safety setpoint upper tolerance. The TS setpoints for the SRVs are not changed. The as-left setpoint tolerances are not changed by the proposed amendment and remain at ±3%.

The margin of safety is established through the design of the plant structures, systems, and components, the parameters within which the plant is operated, and the establishment of the setpoints for the actuation of equipment relied upon to respond to an event. The proposed change from -3% to -5% for the SRV safety setpoint lower tolerance does not significantly impact the condition or performance of structures, systems, and components relied upon for accident mitigation.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, Energy Northwest concludes that the proposed change from -3% to -5% for the SRV safety setpoint lower tolerance does not involve a significant

hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

5.4 Conclusion

Based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be adverse to the common defense and security or the health and safety of the public.

6. ENVIRONMENTAL CONSIDERATION

Title 10 of CFR 51.22(c)(9) identifies certain licensing and regulatory actions, which are eligible for categorical exclusion from the requirement to perform an environmental assessment. A proposed amendment to an operating license for a facility does not require an environmental assessment if operations of the facility in accordance with the proposed amendment would not: (1) involve a significant hazards consideration; (2) result in a significant change in the types or significant increase in the amounts of any effluents that may be released offsite; or (3) result in a significant increase in individual or cumulative occupational radiation exposure. Energy Northwest has evaluated the proposed change and determined that the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Accordingly, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessments needs to be prepared in connection with the issuance of the amendment. The basis for this determination, using the above criteria, follows:

Basis

As demonstrated in the "No Significant Hazards Consideration" evaluation, the proposed amendment does not involve a significant hazards consideration.

There is no significant change in the operational transients. The proposed change does not involve any physical alteration of the plant. No new or different type of equipment will be installed. The proposed change does not involve any change in methods governing normal plant operation. There is no significant increase in individual or cumulative occupational radiation exposure.

7. REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III

2. Energy Northwest, Columbia Generating Station, Final Safety Analysis Report Amendment 63

3. GE-NE-187-24-0992, "Washington Public Power Supply System Nuclear Project 2 SRV Setpoint Tolerance and Out-of-Service Analysis," Revision 2, July 1993

4. NEDC-32115P, " Washington Public Power Supply System Nuclear Project 2 SAFER/GESTRLOCA Loss-of-Coolant Accident Analysis," Revision 2, July 1993

5. Letter, J. V. Parrish (WPPSS) to NRC, “WNP-2, Operating License NPF-21 Request for Amendment to the Facility Operating License and Technical Specifications to Increase Licensed Power Level from 3323 MWt to 3486 MWt with extended Load Line Limit and a Change in Safety Relief Valve Setpoint Tolerance,” dated July 9, 1993

6. Letter, James W. Clifford (NRC) to J. V. Parrish (WPPSS), “Issuance of Amendment for Washington Public Power Supply System Nuclear Project No. 2 (TAC NOS. M87076 and M88625),” dated May 2, 1995

7. 003N3650-R1, “[Columbia Generating Station] CGS SRV Safety Valve Setting Tolerance Change Supplement to GE-NE-187-24-0992, Rev. 2,” April 5, 2016

8. Letter, Eric R. Oesterle (NRC) to Mark E. Reddemann (Energy Northwest), “Columbia Generating Station – Request for Relief Nos. RG01, RP01, RP02, RP03, RP04, RP05, RP06, RV01, RV02, RV03, and RV04 for the Fourth 10-Year Inservice Testing Interval (TAC Nos. MF3847, MF3848, MF3849, MF3851, MF3852, MF3854, MF3855, MF3856, MF3857, and MF3858),” dated December 9, 2014

9. GE-NE-208-17-0993, “Washington Public Power Supply System Power Nuclear Project 2 Uprate Project NSSS Engineering Report, Rev. 1,” December 1994

10. Letter, Bhalchandra K. Vaidya (NRC) to Timothy S. Rausch (PPL), “Susquehanna Steam Electric Station, Units 1 and 2 – Issuance of Amendments RE: Change to Technical Specifications (TSs) Surveillance Requirements (SRs) 3.4.3.1 to Revise Lower Surveillance Tolerances (TAC NOS. ME5050 and ME5051),” dated November 17, 2011

11. Letter, Michael Webb (NRC) to Paul D. Hinnencamp (Entergy), “River Bend Station – Unit 1 – Issuance of Amendment RE: Modification of the Technical Specification Surveillance Requirements for Safety/Relief Valves (TAC NO. MB5090),” dated February 13, 2003

12. ASME, OM Code, 2004 Edition through 2006 Addenda

13. 0000-0162-2826-R0, “CGS Containment Negative Pressure Transient Analysis,” June 2013

14. Crosby Valve and Gage Company Procedure I-11069, “Instruction Manual for Crosby Style 6xRx10 HB-65-BP Safety Relief Valve for Main Steam Service,” Revision 1

Proposed Technical Specification Markup Pages

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) - ≥ 25% RTP

LCO 3.4.3 The safety function of 12 SRVs shall be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows:	In accordance with the Inservice Testing Program

Number of SRVs	Setpoint (psig)
2	1165 ~~±~~+ 34.9 - 58.2
4	1175 ~~±~~+ 35.2 - 58.7
4	1185 ~~±~~+ 35.5 - 59.2
4	1195 ~~±~~+ 35.8 - 59.7
4	1205 ~~±~~+ 36.1 - 60.2

Following testing, lift settings shall be within ±3%.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.2	Verify each required SRV opens when manually actuated.	24 months

SURVEILLANCE REQUIREMENTS

	SURVEILLANCE	FREQUENCY
SR 3.4.4.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs — Setpoint (psig) 2 — 1165 ±+ 34.9 - 58.2 4 — 1175 +± 35.2 - 58.7 4 — 1185 +± 35.5 - 59.2 4 — 1195 +± 35.8 - 59.7 4 — 1205 +± 36.1 - 60.2 Following testing, lift settings shall be within ±3%.	In accordance with the Inservice Testing Program
SR 3.4.4.2	------------------------------NOTE----------------------------- Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each required SRV opens when manually actuated.	24 months

**Proposed Technical Specification Bases Markup Pages
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BASES

APPLICABLE SAFETY ANALYSES (continued)

From an overpressure standpoint, the design basis events are bounded by the MSIV closure with flux scram event described above. References 3, 4, and 5 discuss additional events that are expected to actuate the SRVs. The analysis described in Reference 5 also assumes that, for certain events (e.g., ECCS performance during a small break LOCA), of the 12 required OPERABLE SRVs, two SRVs with lift setpoints in the lowest two lift setpoint groups are OPERABLE.

SRVs - ≥ 25% RTP satisfy Criterion 3 of Reference 6.

LCO

The safety function of 12 SRVs is required to be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE. The requirements of this LCO are applicable only to the capability of the SRVs to mechanically open to relieve excess pressure when the lift setpoint is exceeded (safety mode). In Reference 2, an evaluation was performed to establish the parametric relationship between the peak vessel pressure and the number of OPERABLE SRVs. The results show that with a minimum of 12 SRVs in the safety mode OPERABLE, the ASME Code limit of 1375 psig is not exceeded. While the analysis assumes the overpressurization event is mitigated by SRVs with the highest setpoints (Ref. 2), the small break LOCA analysis (Ref. 5) assumes two of the 12 required OPERABLE SRVs have lift setpoints in the lowest two lift setpoint groups.

The SRV safety setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in References 3, 4, ~~and~~ 5, and 8 involving the safety mode are based on these setpoints, but also include the additional uncertainties of ~~±~~ + 3% / -5% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded or unacceptable core thermal margins.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.4.3.2

A manual actuation of each required SRV is performed to verify that, mechanically, the valve is functioning properly and no blockage exists in the valve discharge line. This can be demonstrated by the response of the turbine governor valves or bypass valves, by a change in the measured steam flow, or any other method suitable to verify steam flow. If the valve fails to actuate due only to the failure of the solenoid but is capable of opening on overpressure, the safety function of the SRV is not considered inoperable.

The 24 month Frequency was developed based on the SRV tests required by the ASME OM Code (Ref. 7). Operating experience has shown that these components usually pass the Surveillance when performed at the 24 month Frequency. Therefore, the Frequency was concluded to be acceptable from a reliability standpoint.

REFERENCES

1. ASME, Boiler and Pressure Vessel Code, Section III.
2. FSAR, Section 15.2.4.
3. FSAR, Chapter 15.
4. GE-NE-187-24-0992, "WPPSS Nuclear Project 2 SRV Setpoint Tolerance and Out-of-Service Analysis," Revision 2, July 1993.
5. NEDC-32115P, Columbia Generating Station, "SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," Revision 2, July 1993.
6. 10 CFR 50.36(c)(2)(ii).
7. ASME Code for Operation and Maintenance of Nuclear Power Plants.
8. 003N3650-RB, CGS SRV Safety Valve Setting Tolerance Change Supplement to GE-NE-187-24-0992, Rev. 2.

BASES

LCO (continued)

The SRV safety setpoints are established to ensure the ASME Code limit on peak reactor pressure is satisfied. The ASME Code specifications require the lowest safety valve be set at or below vessel design pressure (1250 psig) and the highest safety valve be set so the total accumulated pressure does not exceed 110% of the design pressure for overpressurization conditions. The transient evaluations in References 4, ~~and~~ 5, and 8 involving the safety mode are based on these setpoints, but also include the additional uncertainties of ~~±~~+ 3% / -5% of the nominal setpoint to account for potential setpoint drift to provide an added degree of conservatism.

Operation with fewer valves OPERABLE than specified, or with setpoints outside the ASME limits, could result in a more severe reactor response to a transient than predicted, possibly resulting in the ASME Code limit on reactor pressure being exceeded.

APPLICABILITY

In MODE 1 with THERMAL POWER < 25% RTP and MODES 2 and 3, the specified number of SRVs must be OPERABLE since there may be considerable energy in the reactor core and the limiting design basis transients are assumed to occur. The SRVs may be required to limit peak reactor pressure.

The requirements for SRVs with THERMAL POWER ≥ 25% RTP are discussed in LCO 3.4.3. In MODE 4, decay heat is low enough for the RHR System to provide adequate cooling, and reactor pressure is low enough that the overpressure limit cannot be approached by assumed operational transients or accidents. In MODE 5, the reactor vessel head is unbolted or removed and the reactor is at atmospheric pressure. The SRV function is not needed during these conditions.

ACTIONS

A.1 and A.2

With less than the minimum number of required SRVs OPERABLE, a transient may result in the violation of the ASME Code limit on reactor pressure. If one or more required SRVs are inoperable, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

BASES

REFERENCES

1. FSAR, Section 15.2.4.
2. Columbia Generating Station Calculation NE-02-94-66, Revision 0, November 13, 1995.
3. ASME, Boiler and Pressure Vessel Code, Section III.
4. FSAR, Chapter 15.
5. GE-NE-187-24-0992, "WPPSS Nuclear Project 2 SRV Setpoint Tolerance and Out-of-Service Analysis," Revision 2, July 1993.
6. 10 CFR 50.36(c)(2)(ii).
7. ASME Code for Operation and Maintenance of Nuclear Power Plants.
8. 003N3650-RB, CGS SRV Safety Valve Setting Tolerance Change Supplement to GE-NE-187-24-0992, Rev. 2.

Proposed Technical Specification Clean Pages

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.3 Safety/Relief Valves (SRVs) - ≥ 25% RTP

LCO 3.4.3 The safety function of 12 SRVs shall be OPERABLE, with two SRVs in the lowest two lift setpoint groups OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 25% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required SRVs inoperable.	A.1 Reduce THERMAL POWER to < 25% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs / Setpoint (psig) 2 — 1165 + 34.9 / - 58.2 4 — 1175 + 35.2 / - 58.7 4 — 1185 + 35.5 / - 59.2 4 — 1195 + 35.8 / - 59.7 4 — 1205 + 36.1 / - 60.2 Following testing, lift settings shall be within ±3%.	In accordance with the Inservice Testing Program

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.2	Verify each required SRV opens when manually actuated.	24 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.4.1	Verify the safety function lift setpoints of the required SRVs are as follows: Number of SRVs — Setpoint (psig) 2 — 1165 + 34.9 / - 58.2 4 — 1175 + 35.2 / - 58.7 4 — 1185 + 35.5 / - 59.2 4 — 1195 + 35.8 / - 59.7 4 — 1205 + 36.1 / - 60.2 Following testing, lift settings shall be within ±3%.	In accordance with the Inservice Testing Program
SR 3.4.4.2	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after reactor steam pressure and flow are adequate to perform the test. --- Verify each required SRV opens when manually actuated.	24 months