and to provide needed clarification. The applicable TS Bases sections would also be changed to reflect the proposed changes, correct previous errors identified during the licensee's review of the TS, eliminate redundant information, and expand the TS Bases to discuss the new requirements for the automatic isolation of the steam generator blowdown.

Specifically, the proposed changes would modify TS 2.1.1, ''Safety Limits—Reactor Core,'' TS 2.2.1, ''Limiting Safety System Settings—Reactor Trip Setpoints,'' TS 3.3.1.1, ''Instrumentation—Reactor Protective Instrumentation'' TS 3.3.2.1, ''Instrumentation—Engineered Safety Features Actuation System Instrumentation,'' and would add a new TS 3.7.1.8, ''Plant Systems—Steam Generator Blowdown Isolation Valves.'' As previously noted, the applicable TS Bases sections will be updated to reflect the proposed changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to correct the maximum reactor power level from 112% to 111.6% is consistent with the maximum high power trip setpoint of 106.6%, plus 5% uncertainty, currently used in the safety analyses. This does not change the Technical Specification required high power reactor trip setpoint. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the trip setpoints and allowable values for the Reactor Protection System (RPS) trips on high pressurizer pressure, high containment pressure, low steam generator pressure, and low steam generator level are the result of revisions to the instrument loop uncertainty and setpoint calculations. These calculations were revised to incorporate calculation methodology changes, analytical limit changes, correct errors identified, and to include the effects of a harsh environment (pressure, temperature, and radiation), where appropriate. The proposed setpoints and allowable values will ensure a reactor trip signal is generated at, or before the analytical limits used in the respective accident analyses are reached. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the trip setpoints and allowable values for the Engineered Safety Features Actuation System (ESFAS) actuations on low pressurizer pressure, high containment pressure, low steam generator pressure, low refueling water storage tank level, and low steam generator level are the result of revisions to the instrument loop uncertainty and setpoint calculations. These changes were revised to incorporate calculation methodology changes, analytical limit changes, correct errors identified, and to include the effects of a harsh environment (pressure, temperature, and radiation), where appropriate. The proposed setpoints and allowable values will ensure an ESF [engineered safety feature] actuation signal is generated at, or before the analytical limits used in the respective accident analyses are reached. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to add Technical Specification requirements for the steam generator blowdown isolation valves will provide additional assurance that the automatic isolation of steam generator blowdown will occur as assumed in the loss of main feedwater accident analysis. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the value of steam generator pressure when the steam generator low pressure reactor trip can be bypassed (from 780 psia to 800 psia) will reduce the range of plant operation when this trip is required to be available. However, this will not affect the range of plant operation when this RPS trip is required to be operable. This RPS trip is required in Modes 1 and 2. The expected steam generator pressure during a reactor startup (entry into Mode 2) is approximately 900 psia, which corresponds to a Reactor Coolant System (RCS) temperature of approximately 532°F. The proposed change will require the bypass to be automatically removed prior to exceeding a steam generator pressure of 800 psia. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the value of pressurizer pressure (from 1750 psia to 1850 psia) when the pressurizer low pressure ESF actuations (SIAS, CIAS, and EBFAS) [safety injection actuation system, containment isolation actuation system, and enclosure building filtration actuation system] can be blocked will reduce the range of plant operation when these functions are required to be available. However, since the plant would normally be in Mode 3 when pressurizer pressure is in this range, automatic actuation of these ESF functions on high containment pressure, as well as manual actuation, is required to be operable. In addition, the plant would not normally maintain pressurizer pressure between 1750 psia and 1850 psia. Therefore, since automatic actuation of these ESF functions on high containment pressure, as well as manual actuation, should be operable, and the time the plant will operate between 1750 psia and 1850 psia is small, the ESFAS will continue to function as before. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed change to the value of steam generator pressure (from 600 psia to 700 psia) when the steam generator low pressure ESF actuation (main steam line isolation) can be blocked will reduce the range of plant operation when this function is required to be available. However, since the plant would be in Mode 3 when steam generator pressure is in this range (RCS temperature of approximately 486°F to 503°F), automatic actuation of this ESF function on high containment pressure, as well as manual actuation, is required to be operable. In addition, the plant would not normally maintain steam generator pressure between 600 psia and 700 psia. Therefore, since automatic actuation of this ESF function on high containment pressure, as well as manual actuation, should be operable, and the time the plant will operate between 600 psia and 700 psia is small, the ESFAS will continue to function as before. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The minor editorial and non-technical changes to correct spelling errors, correct a capitalization error, add page amendment numbers, add the specific plant parameter (steam generator pressure) to use if an RPS or ESF function can be bypassed, change the value of the parameter (pressurizer pressure) used in action statements, and a ''[less than or equal to]'' symbol, change ''value'' to ''setpoint,'' and update the index will have no effect on plant operation. These changes will not result in any technical changes to the Millstone Unit No. 2 Technical Specifications. There will be no adverse effect on any design basis accident previously evaluated or on any equipment important to safety. Therefore, the proposed change will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes to the Technical Specification Bases will incorporate the RPS and ESFAS setpoint changes, correct errors, eliminate redundant information, and expand the Bases to discuss the new requirements for steam generator blowdown isolation. These changes will have no effect on equipment operation. There will be no adverse effect on any design basis accident

previously evaluated or on any equipment important to safety. Therefore, the proposed changes will not result in a significant increase in the probability or consequences of an accident previously evaluated.

The proposed changes have no adverse effect on any of the design basis accidents previously evaluated and have no adverse effect on how the RPS and ESFAS function to mitigate the consequences of design basis accidents. Therefore, the license amendment request does not impact the probability of an accident previously evaluated nor does it involve a significant increase in the consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

The proposed changes will not alter the plant configuration (no new or different type of equipment will be installed) or require any new or unusual operator actions. They do not alter the way any structure, system, or component functions and do not alter the manner in which the plant is operated. The proposed changes do not introduce any new failure modes. Therefore, the proposed changes will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Involve a significant reduction in a margin of safety.

The proposed changes will correct the maximum reactor power level specified; change RPS trip setpoints, allowable values, and bypass setpoints; change ESFAS trip setpoints, allowable values, and block setpoint changes; add a new Technical Specification and additional requirements associated with the automatic isolation of steam generator blowdown; and make various minor editorial and non-technical changes. There will be no adverse effect on equipment important to safety. The RPS and ESFAS will continue to function as designed to mitigate the consequences of design basis accidents. Therefore, there will be no significant reduction of the margin of safety as defined in the Bases for the Technical Specifications affected by the proposed changes.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Local Public Document Room location: Learning Resources Center, Three Rivers Community-Technical College, 574 New London Turnpike, Norwich, Connecticut, and the Waterford Library, ATTN: Vince Juliano, 49 Rope Ferry Road, Waterford, Connecticut.

Attorney for licensee: Lillian M. Cuoco, Esq., Senior Nuclear Counsel, Northeast Utilities Service Company, P.O. Box 270, Hartford, Connecticut.

NRC Deputy Director: Phillip F. McKee.

Pennsylvania Power and Light Company, Docket No. 50–387, Susquehanna Steam Electric Station, Unit 1, Luzerne County, Pennsylvania

Date of amendment request: June 19, 1998.

Description of amendment request: The amendment to Unit 1 Technical Specifications (TS) involves the addition of a new section entitled ''Oscillation Power Range Monitoring (OPRM) Instrumentation'' and revisions to Section 3.4.1 ''Recirculation Loops Operating'' to remove the specifications related to thermal power stability which will not be required after the installation of the OPRM instrumentation. Unit 1 is currently operating under Interim Corrective Actions (ICAs) defined in TS 3.4.1 that specify restrictions on plant operation and actions by operators in response to instability events. The OPRM system provides an automatic long-term solution to the instability issue and eases the burden on the operator.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

This proposal does not involve an increase in the probability or consequences of an accident previously evaluated.

The OPRM most directly affects the APRM and LPRM portions of the Power Range Neutron Monitoring system. Its installation does not affect the operation of these sub-systems. None of the accidents or equipment malfunctions affected by these sub-systems are affected by the presence or operation of the OPRM.

The APRM channels provide the primary indication of neutron flux within the core and respond almost instantaneously to neutron flux changes. The APRM Fixed Neutron Flux-High function is capable of generating a trip signal to prevent fuel damage or excessive reactor pressure. For the ASME overpressurization protection analysis in FSAR Chapter 5, the APRM Fixed Neutron Flux-High function is assumed to terminate the main steam isolation valve closure event. The high flux trip, along with the safety/relief valves, limit the peak reactor pressure vessel pressure to less than the ASME Code limits. The control rod drop accident (CRDA) analysis in Chapter 15 takes credit for the APRM Fixed Neutron Flux-High function to terminate the CRDA. The Recirculation Flow Controller Failure event (pump runup) is also terminated by the high neutron flux trip. The APRM Fixed Neutron Flux-High function is required to be OPERABLE in MODE 1 where the potential consequences of the analyzed transients could result in the Safety Limits (e.g., MCPR and Reactor pressure) being exceeded.

The installation of the OPRM equipment does not increase the consequences of a malfunction of equipment important to safety. The APRM and RPS systems are designed to fail in a tripped (fail safe) condition; the OPRM will have no affect on the consequence of the failure of either system. An inoperative trip signal is received by the RPS any time an APRM mode switch is moved to any position other than Operate, an APRM module is unplugged, the electronic operating voltage is low, or the APRM has too few LPRM inputs. These functions are not specifically credited in the accident analysis, but are retained for the RPS as required by the NRC approved licensing basis.

The OPRM allows operation under current operating conditions presently restricted by the current Technical Specifications by providing automatic suppression functions in the area of concern in the event an instability occurs. The consequences of any accident or equipment malfunction are not increased by operating under those conditions. Although protected by the OPRM from thermal-hydraulic core instabilities above 30% core power, operation under natural core recirculation conditions is not allowed. No accidents or transients of a type not analyzed in the FSAR are created by operating under these conditions with the protection of the OPRM system.

This change does not increase the probability of an accident as previously evaluated. The OPRM is designed and installed to not degrade the existing APRM, LPRM, and RPS systems. These systems will still perform all of their intended functions. The new equipment is tested and installed to the same or more restrictive environmental and seismic envelopes as the existing systems. The new equipment has been designed and tested to the electromagnetic interference (EMI) requirements of Reference 2, which assures correct operation of the existing equipment. The new system has been designed to single failure criteria and is electrically isolated from equipment of different electrical divisions and from non-1E equipment. The electrical loading is within the capability of the existing power sources and the heat loads are within the capability of existing cooling systems. The OPRM allows operation under operating conditions presently forbidden or restricted by the current Technical Specifications. No other transient or accident analysis assumes these operating restrictions.

Based upon the analysis presented above, PP&L concludes that the proposed action does not involve an increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

This proposal does not create the probability of a new or different type of accident from any accident previously evaluated. The OPRM system is a monitoring and accident mitigation system that cannot create the possibility for an accident.