SOUTHERN CALIFORNIA
EDISON
An EDISON INTERNATIONAL℠ Company

Dwight E. Nunn
Vice President

May 25, 2001

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: **Docket Nos. 50-361 and 50-362**
Proposed Change Number NPF-10/15-514
Increase in Reactor Power to 3438 MWt
San Onofre Nuclear Generating Station, Units 2 and 3

References:

1. SCE to NRC letter dated April 3, 2001, Subject: Proposed Change Number NPF-10/15-514 Increase in Reactor Power to 3438 MWt, San Onofre Nuclear Generating Station Units 2 and 3
2. SCE to NRC letter dated May 11, 2001, Subject: Proposed Change Number NPF-10/15-514 Increase in Reactor Power to 3438 MWt, San Onofre Nuclear Generating Station Units 2 and 3

Gentlemen:

This letter provides responses to NRC requests for additional information (RAIs) concerning the Southern California Edison (SCE) request to increase the reactor power to 3438 MWt at San Onofre Units 2 and 3, Amendment Applications 207 and 192, Proposed Change Number 514 (Reference 1).

Enclosure 1 provides information requested in an April 20, 2001 telephone call. Reference 2 provided SCE responses to earlier NRC requests for additional information from an April 24, 2001 telephone call. Enclosure 2 responds to an item remaining (from Reference 2).

If you have any questions regarding these amendment applications, please contact me or Mr. Jack L. Rainsberry (949) 368-7420.

Sincerely,

P. O. Box 128
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A001

Enclosures

cc: E. W. Merschoff, Regional Administrator, NRC Region IV
C. C. Osterholtz, NRC Senior Resident Inspector, San Onofre Units 2 and 3
J. E. Donoghue, NRC Project Manager, San Onofre Units 2 and 3
S. Y. Hsu, Department of Health Services, Radiologic Health Branch

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 2 of the San Onofre Nuclear Generating Station	))))))	Docket No. 50-361 Amendment Application No. 207

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit information in support of Amendment Application No. 207. This information consists of responses to NRC requests for additional information on Proposed Change No. NPF-10-514 to Facility Operating License NPF-10. Proposed Change No. NPF-10-514 is a request to revise the Facility Operating License by increasing the licensed power for operation.

Subscribed on this 25th day of May, 2001.

Respectfully submitted,
SOUTHERN CALIFORNIA EDISON COMPANY

By: [signature]
Dwight E. Nunn
Vice President

State of California
County of San Diego
On 5/25/01 before me, Mariane Sanchez, personally appeared Dwight E. Nunn personally known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same in his authorized capacity, and that by his signature on the instrument the person, or the entity upon behalf of which the person acted, executed the instrument.
WITNESS my hand and official seal.

Signature [signature]

MARIANE SANCHEZ
Commission #1196482
Notary Public - California
San Diego County
My Comm. Expires Oct 14, 2002

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

Application of SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. for a Class 103 License to Acquire, Possess, and Use a Utilization Facility as Part of Unit No. 3 of the San Onofre Nuclear Generating Station	))))))	Docket No. 50-362 Amendment Application No. 192

SOUTHERN CALIFORNIA EDISON COMPANY, ET AL. pursuant to 10 CFR 50.90, hereby submit information in support of Amendment Application No. 192. This information consists of responses to NRC requests for additional information on Proposed Change No. NPF-15-514 to Facility Operating License NPF-15. Proposed Change No. NPF-15-514 is a request to revise the Facility Operating License by increasing the licensed power for operation.

Subscribed on this 25th day of May, 2001.

Respectfully submitted,
SOUTHERN CALIFORNIA EDISON COMPANY

By: ______________________
Dwight E. Nunn
Vice President

State of California
County of San Diego
On 5/25/01 before me, Mariane Sanchez, personally appeared Dwight E. Nunn, personally known to me to be the person whose name is subscribed to the within instrument and acknowledged to me that he executed the same in his authorized capacity, and that by his signature on the instrument the person, or the entity upon behalf of which the person acted, executed the instrument.
WITNESS my hand and official seal.

Signature ______________________

MARIANE SANCHEZ
Commission #1196482
Notary Public - California
San Diego County
My Comm. Expires Oct 14, 2002

Enclosure 1

Item 1. In Section 3.1 of the submittal, SCE indicated that the key design parameters for this amendment request fell at or between the current operating conditions (reduced Tcold) and the original plant design. In support of the stated bounding conditions, please provide a comparison of the key design parameters (Reactor Coolant System (RCS) pressure, RCS hot leg and cold leg temperatures, steam generator (SG) pressure, and SG outlet temperature, feedwater (FW) temperature and flow rate) for the proposed power uprate, the current operating (reduced Tcold) and the original plant design conditions.

Response:

The requested parameters for the San Onofre Nuclear Generating Station (SONGS) Units 2 and 3 are provided in the following Table 1:

TABLE 1 - Key Design Parameters for Steady State Hot Full Power Operation

Parameter	Original Design (1)	Current Operating Condition (1)	Projected Post-Power Uprate (1)	Ranges Evaluated in the UFSAR Chapter 15 Accident Analysis Including Instrument Uncertainty
Pressurizer Pressure	2250 psia	2250 psia	2250 psia	2000 - 2300 psia
RCS Thot	611°F	596°F	599°F (2)	590 - 617°F
RCS Tcold	553°F	540°F	542°F (2)	533 - 560°F
Steam Generator Pressure	900 psia	806 psia	816 psia	780 - 900 psia
Steam Generator Outlet Temperature	532°F	519°F	521°F	515 - 532°F
Main Feedwater Temperature	444.8°F	441.5°F	442°F	>400.0°F
Main Feedwater Flow Rate	15.24×10^6 lbm/ hour	15.09×10^6 lbm/hour	15.3×10^6 lbm/hour	14.16 E6 lbm/hr to 21.9 E6 lbm/hr

(1) Nominal Values are representative and may vary by loop and unit.

(2) The temperatures shown are estimates for mechanical calculations. The actual maximum temperatures will be evaluated by the SONGS Steam Generator Program.

Item 2. In Section 3.3.3, SCE evaluated the reactor internals and stated that with little or no increase in thermal design flow and changes in the RCS temperatures there will be little or no changes in the boundary conditions experienced by the reactor internals components. SCE also indicated that increases in core thermal power will slightly increase nuclear heating rates in the reactor vessel internals, but the internals remain within the design capability of the system analysis. Please provide a summary of the SCE evaluation to show that the flow and temperature increase in the reactor internals are bounded by the current design basis analysis of the reactor internals. Also, please confirm that there is no increase in the potential for flow induced vibration for the reactor internal components.

Response:

Reactor Vessel Internals Evaluation

Reactor Vessel Internals (RVI) were re-evaluated for structural integrity, component design metal temperatures, nuclear heating, and hydraulic loading for the Tcold reduction amendment applications (Reference 1). The re-evaluation consisted of comparing hydraulic loads, pressures and temperatures resulting from reducing Tcold temperature against the hydraulic loads, pressures and temperatures used in the original design analysis for these components. Section 3.3.3 of the Uprate amendment applications (Reference 2) provided a general discussion of the uprate conditions on the reactor internals.

Pressure/Loads - Reactor Vessel Internals (RVI)

The RCS operating pressure is unaffected by the Tcold reduction or power uprate change. Reducing the temperature of the primary coolant has the potential to increase the loads on the reactor internals and fuel during a pipe break because of an increase in water density. The slightly higher Tcold temperature resulting from the power uprate change would be associated with a slight decrease in the water density and, as such, the power uprate change is bounded by the evaluation performed previously for the Tcold reduction.

Temperature - RVI Component Design Metal Temperature

Maximum core power of 3458 MWt (3390 MWt + 2% for uncertainty) was used in evaluating the RVI component design metal temperatures for the Tcold reduction change. The evaluation showed adequate design metal temperatures for RVI Components for the Tcold reduction change. Since the nuclear heat rates have been bounded for the uprate, the analyses performed for the Tcold reduction are bounding.

Flow - Reactor Vessel Internals (RVI)

Tcold reduction re-evaluated the hydraulic loads for RVI components at mechanical design conditions of 120% of design flow and 500°F. Vortex shedding frequency was also included in the evaluation for the affected RVI components at the reduced Tcold temperature. Since the changes in RCS flow are insignificant compared to the design parameters considered, the analysis performed for Tcold reduction bounds the power uprate condition.

Item 3. In Section 3.4.2, SCE evaluated the structural integrity of the SGs and indicated that the existing structural and fatigue analysis of the SGs in SONGS Units 2 and 3 was reviewed by comparing the uprate condition to the current design basis analysis of record to determine if the analysis of record remains bounding. Please provide a summary of comparison for each of the design parameters (i.e., the primary and secondary system pressures and pressure differentials) between the current design basis and the power uprate condition. Also, please confirm that there is no increase in the potential for the flow induced vibration and fatigue usage for the U-bend tubes for the power uprate.

Response:

1. Structural Integrity of the Steam Generators

The steam generator (SG) original design report (Calculations 1 and 2) and the addendum addressing the Tcold reduction (Calculation 3) were reviewed to evaluate the impact of the power uprate. The Tcold reduction report identified the SG components requiring an additional fatigue evaluation, and a fatigue analysis was performed for these components at several key locations. The design parameters included the cold leg temperature and the secondary side pressure corresponding to several identified plant transients. Revised usage factors were calculated, and were found acceptable for all SG components at all key locations. In the original design report, the steam generator tubes cumulative usage factor (CUF) was calculated to be 0. Furthermore, the tube stresses and vibration displacements calculated in the degraded eggcrate analysis (Calculation 4) were found to be negligible. This analysis was conducted using actual plant operating parameters. The steam generator tubes were not identified for additional fatigue evaluation for the Tcold reduction. Under the projected power uprate conditions, any change in CUF for the tubes will be insignificant when compared to the CUF limit of 1.0.

The comparison provided in Table 1 (see response to Item 1 above) shows that the pressure and temperature associated with the projected post power uprate have intermediate values lying between the original design and the post Tcold reduction values. Therefore, results of the current analyses are considered bounding.

2. Flow Induced Vibrations

Significant degradation of SONGS Unit 3 SG periphery eggcrate supports was observed during the Cycle 9 refueling outage. This condition was analyzed (Calculation 4) for its effect on steam generator integrity. Tubes with two or more consecutive degraded supports were staked and plugged. Thus, the SG tubes most susceptible to flow induced vibration (FIV) are:

(a) Tubes with one eggcrate support uncredited.
(b) Tubes with alternate eggcrate supports uncredited.

A detailed flow induced vibration (FIV) analysis was performed as part of the Tcold reduction at SONGS for all SG tubes remaining in service, including the limiting configurations (a) and (b) above. The stability ratio (SR) of the worst case tube was evaluated based on an increase in the average secondary side fluid velocity of 4%. As a result, the value of the maximum SR increased from 0.64 to 0.75, which is acceptable since it is below the acceptable limit of 1.0. The increased power level will cause a slight decrease in the recirculation ratio (because of increased steam flow); however, total flow in the steam generator will not change significantly. The fluid in the tube bundle will have lower densities but higher velocities. Based on studies Westinghouse has performed for steam generators identical to San Onofre, these changes will cause axial velocities in the tube bundle to increase by a maximum of 4% to 5%, or similar to the change that was seen during the Tcold reduction analysis. It is therefore anticipated that the SR for the worst tube row will increase by a similar amount to a value of approximately 0.88. Since this value is below the acceptance criterion of 1.0, the power uprate will not result in unacceptable tube vibration. During preparation of this response we noticed that Section 3.4.2.4 of the power uprate amendment requests stated that feedwater flow impact was bounded by previous evaluations. Since the power uprate feedwater flow is slightly increased over the Tcold reduction, it is not bounded by the Tcold reduction evaluation. However, the conclusion is the same, as shown above, the increased feedwater flow impact remains below acceptance criteria.

Calculations Reviewed in the Above Response

1. CENC-1272, "Analytical Report for Southern California Edison San Onofre Unit 2 Steam Generators"

2. CENC-1298, "Analytical Report for Southern California Edison San Onofre Unit 3 Steam Generators"

3. Calculation No. SO23-915-C220, Revision 1 (Westinghouse Calculation number DR-SONGS-9449-1207, Revision 1), "Addendum to CENC-1272 and CENC-1298, Analytical Reports for Southern California Edison SONGS Units 2 and 3 Steam Generators"

4. Calculation No. SO23-915-208, Revision 1, (Westinghouse Calculation number A-SONGS-9449-1198, Revision 1), "Evaluation of Southern California Edison SONGS Unit 3 Steam Generators with Degraded Eggcrates"

Item 4. Please provide a summary of evaluations for the reactor vessel, pressurizer, and NSSS piping. The information should include the existing minimum margin in stress and CUF (cumulative usage factor) which will accommodate the slight changes for the 1.42 percent power uprate or to show that the component design basis temperatures or temperature differentials are bounding for the power uprate condition.

Response:

The following is a summary of structural evaluations for the effects of power uprate:

Objective:

Determine the impact of the planned 1.42% power uprate on the structural and pressure boundary integrity of the SONGS Unit 2 and 3 Reactor Coolant Systems (RCSs). The equipment considered in this evaluation consists of the RCS major components (i.e., Reactor Vessel, Steam Generators, Reactor Coolant Pumps, Pressurizer, and Main Coolant Loop (MCL) Piping) and the major component Supports.

Background:

Refer to Table 1 (see response to Item 1 above) for a comparison of the expected operating parameters to original design and post Tcold Reduction operating parameters.

The original design basis RCS structural integrity analyses performed by the NSSS vendor considered three types of loadings: steady state and transient normal operating loads, seismic loads, and Loss Of Coolant Accident (LOCA) loads. Pressure, temperature, and weight effects were statically analyzed for steady state normal operation. Normal operating transient analyses considered changes in temperature and pressure over time during pre-defined plant operating events. Seismic analyses considered loads due to operating basis and safe shutdown earthquakes. LOCA analyses were performed for a number of pipe break events. After determining the RCS loads resulting from these events, ASME Code stress and fatigue analyses were performed to determine results at critical locations throughout the system. These results were compared to the Code allowables to demonstrate the structural integrity of the equipment and supports.

SCE has completed engineering evaluations justifying a Tcold reduction of 13°F from the original full power operating condition of 553°F (Reference 1). This change in Tcold and the corresponding changes in other plant operating parameters were examined for their effect on the RCS Analyses of Record (AOR) for stresses and Cumulative Usage Factors (CUFs).

It was concluded in the Tcold Reduction amendment applications that the calculated stress values and fatigue factors at all critical primary system component and support locations remained below the allowable limits determined in the analysis of record (AOR). In the case of the SGs, some transients defined in the original fatigue evaluations produced larger stress ranges under Tcold conditions, which resulted in higher CUFs. In all cases, however, the stresses at critical locations continued to meet ASME Code criteria.

Analytical Approach:

The projected post-power uprate operating parameters of the primary system shown in Table 1 were reviewed and compared to both the original design conditions and the operating conditions associated with Tcold Reduction. All parameters corresponding to the proposed power uprate fall between the primary system parameters associated with the original design and the Tcold conditions. Consequently, the effects of the proposed power uprate are bounded by previous analyses.

Please refer to the response to Item 3 for a more detailed discussion of Steam Generator structural integrity. The following applies to the remaining RCS major components and supports.

Assessment of Effects on Non-Faulted Condition Stresses and CUFs:

Reactor Vessel, Reactor Coolant Pumps, and MCL Piping:

As shown in Table 1, all operating parameters associated with these primary system components are bounded by the original design basis and the Tcold reduction conditions. Therefore, the effect of the proposed power uprate on the associated transients is also bounded by previous analyses and assessments, leading to the conclusion that all secondary thermal stresses previously determined for these components and component supports bound the proposed power uprate conditions. The primary stresses due to pressure resulting from an operating basis earthquake would not change for the proposed power uprate; therefore, the previously determined CUFs continue to be bounding.

Pressurizer:

All operating parameters are unchanged for the Pressurizer; therefore, the Pressurizer would be unaffected by the proposed power uprate.

Assessment of Effects on Faulted Condition Stresses:

The LOCA contribution to the faulted loads (i.e., the combination of normal operating, safe shutdown earthquake, and LOCA loads) increases as Tcold decreases, because the rapid depressurization at the break and the resulting blowdown become more severe and produce higher system loads. The Tcold reduction analysis, however, demonstrated that the implementation of Leak-Before-Break methodology for the RCS more than compensates for the increased LOCA loads resulting from the lowered Tcold operating temperature, and produces acceptable Faulted condition stresses and CUFs. Since the projected Tcold temperature for the proposed power uprate (542°F) is greater than the Tcold temperature associated with Tcold Reduction (540°F), LOCA loads and, therefore, any Faulted Condition stresses remain bounded and acceptable.

Conclusions:

The shift in some operating parameters, relative to the two previously analyzed sets of operating parameters defining the original design basis and the post Tcold Reduction design basis, will produce some minor changes in the RCS stresses and CUFs thus far determined for SONGS Units 2 and 3. However, these changes in stresses and CUFs fall within the ranges already determined by the previous analyses and assessments and remain acceptable. Therefore, no further assessments are required to demonstrate that the structural and pressure boundary integrity of the SONGS Unit 2 and 3 RCSs will be maintained for the proposed 1.42% power uprate.

Item 5. In Section 3.6.6, SCE indicated that the motor-operator valve (MOV) program at SONGS was set up in such a way that setpoints were established and MOVs were tested to demonstrate their capability to perform their safety related function. MOV setpoint evaluations include several conservatisms, and small changes in the system operating pressure are not expected to impact the operation of these MOVs. SCE also indicated that the proposed increase in flow rate has no significant impact on the operation of gate and globe MOVs since the expected changes in the differential pressure are insignificant and that a small increase in flow rate would increase the valve sizing coefficients slightly for butterfly valves. Please confirm that the existing design basis analysis bounds the 1.42 percent power uprate condition associated with the system pressure, temperature, flow rate, and pressure and temperature differentials. Also, please confirm that there will be no impact on the plant safety related valves including air-operated and motor-operated valves and Generic Letter (GL) 89-10 MOV program, and that there are no changes in the post loss-of-coolant-accident (LOCA) conditions associated with GL 95-07 and GL 96-06, following the 1.42 percent power uprate.

1. Operational Basis Calculations

Several Operational Basis Calculations (OBC) (Calculations 3-5) were prepared to provide the necessary design inputs required for performing the set point calculations for all the Motor Operated Valves (MOVs) including the Generic Letter (GL) 89-10 valve population at SONGS. These design inputs include the bounding differential pressure the MOV could be subjected to under normal and accident conditions. An evaluation was made in the setpoint calculations, based on the bounding differential pressure and flow conditions, to ensure that the actuator capability exceeds the valve requirements under the bounding conditions, and the valve will perform its safety function. The bounding conditions include Reactor Thermal Power (RTP) of 102%. This bounds the 1.42% power uprate condition.

2. Generic Letter 95-07

2.1 Pressure Locking Calculations

All MOVs included in the GL 89-10 valve population were screened to determine their susceptibility to Pressure Locking (PL) and Thermal Binding (TB) as required by GL 95-07 (Calculations 1 and 2). Two valve groups, including two valves designs, were identified as potentially susceptible to PL: Shut Down Cooling (SDC) suction isolation valves manufactured by WKM, and Containment Spray system valves manufactured by Target Rock. The evaluation below addresses the impact of the reactor power uprate on these two valve designs.

(a) SDC WKM valves: These double disc gate valves are normally closed to provide isolation for the reactor coolant system. The maximum bonnet pressure in these valves is limited by the lift off pressure of the relief device installed in the segment. Consequently, the maximum bonnet pressure is not affected by the reactor power upgrade. Furthermore, the valve differential pressure is not impacted since there is no increase in the Reactor Coolant System (RCS) pressure. Therefore, there is no impact on the existing PL evaluation of the SDC WKM valves.

(b) Target Rock valves: A total of six (6) of these double disc gate valves exist per unit. Two valves per unit are part of the containment spray system and were evaluated based on the upstream pressure supplied by the containment spray pump discharge head, which is not affected by the subject reactor power uprate. The remaining four valves per unit are part of the chemical and volume control system, and are not impacted by the subject power uprate.

2.2 Thermal Binding

All MOVs included in the GL 89-10 valve population were screened to determine their susceptibility to Thermal Binding (TB) as required by GL 95-07 (Calculation 6). Two valves per unit were identified as potentially susceptible to TB. The following corrective actions were implemented to ensure that the safety function of these valves is not impacted by TB:

(a) Procedural modifications to the shutdown cooling system operation to limit the valve closing temperature.
(b) Gear changes to the motor operator of the valve to increase the output thrust capability.

These actions are not impacted by the subject power uprate. Accordingly, the reactor power uprate does not impact the TB evaluation of these valves.

3. Generic Letter 96-06

An evaluation was performed to ensure containment integrity under a Design Basis Accident (DBA) in response to GL 96-06. All containment penetrations were addressed in this evaluation, and calculations were performed as necessary to ensure that the thermal expansion of trapped fluid under a DBA will not result in a loss of containment integrity. Two DBAs were considered: Loss of Coolant Accident (LOCA) and Main Steam Line Break (MSLB). Conditions inside containment, for both DBAs, assume 102% reactor power. Therefore, the reactor uprate is bounded, and the existing GL 96-06 evaluation is not impacted.

4. Air Operated Valves (AOVs)

Atmospheric dump valves and main steam isolation bypass valves are addressed in Sections 3.5.1.2 and 3.5.1.3 of the power uprate amendment applications (Reference 2), respectively. Other AOVs at SONGS, including safety related and non-safety related valves, are set up based on the system design conditions plus some tolerances. Since the design conditions bound the power uprate conditions, the design bases for the AOV population are not affected by power uprate.

Calculations Reviewed in the Above Response

1. Calculation No. A-95-NM-MOV-PL/TB-002,"GL 95-07 Pressure Locking and Thermal Binding Screening Evaluation."

2. Calculation No. A-96-NM-MOV-PL/TB-003, Revision 1,"GL 95-07 Pressure Locking and Thermal Binding Performance Evaluation."

3. Calculation No. M-8910-1218-OB-001,"Operational Basis Calculation for BAMU Valve HV-9247."

4. Calculation No. M-8910-1218-OB-002,"Operational Basis Calculation for Boric Acid Gravity Feed Valves HV-9235 and HV-9240."

5. Calculation No. M-8910-1218-OB-003, "Operational Basis Calculation for VCT Outlet and RWST to Charging Valves LV-0227B and LV-0227C."

6. Calculation No. M-DSC-343, Revision 0,"Thermal Over Pressurization – Containment Penetrations."

Enclosure 1 References

1. SCE Amendment Applications 179 and 165 for San Onofre Units 2 and 3 RCS Temperature Reduction, dated June 19, 1998, as supplemented by letters dated December 4, 1998, and January 13, 1999.

2. SCE letter to the NRC dated 4/3/2001, Docket Nos. 50-361 and 50-362, Proposed Change Number NPF-10/15-514, Increase in Reactor Power to 3438 MWt, San Onofre Nuclear Generating Station, Units 2 & 3.

Enclosure 2

Item 1. Provide a description of the process for controlling power level when the Crossflow is out of service and how corresponding plant instrument drift values are determined. Describe how "good"and "bad" flags are assigned to correction factors.

Response:

The process for controlling power level when the Crossflow is out of service is:

If the CROSSFLOW system were out of service, plant operation would continue using the most recently generated correction factors for a designated period of time, i.e., allowed outage time. Out of service is defined as when the CROSSFLOW system is unable to provide good quality flow measurements. Allowed outage time is discussed below. If the CROSSFLOW system remains out of service in excess of the allowed outage time, reactor power will be reduced to the currently licensed rated thermal power of 3390 MWt.

Corresponding plant instrument drift values determination:

The instrument drift is quantified by statistical analysis of historical instrument loop calibration data and is a component of the uncertainty calculations for these instruments. The currently allowed outage time of 31 days is calculated assuming the worst case drift, i.e., feedwater flow. The drift is combined with the CROSSFLOW flow uncertainty and a new total secondary calorimetric power uncertainty is recalculated. The total secondary calorimetric power uncertainty results will be less than the reported uncertainty of 0.58%. The allowed outage time may be adjusted with future drift data.

Description of how "good"and "bad" flags are assigned to correction factors:

The new CROSSFLOW system more accurately determines feedwater flow, steam flow, and blowdown flow than the existing instrumentation. The CROSSFLOW system generates a "correction factor" for the existing feedwater and steam flow input signals using the more accurate flow measurement information. The CROSSFLOW system statistically combines successive readings to increase accuracy of the flow readings. Flow reading accuracy is validated by the CROSSFLOW software. If the expected accuracy as described in the reference is achieved, a "good" flag is generated; if the flow signals deviate from the expected accuracy, a "bad" flag is generated.

What is the increased uncertainty from using CROSSFLOW correction factors applied to the existing flow instrumentation versus using CROSSFLOW measurements directly?

Under automatic update conditions the CROSSFLOW calibration factors for process feedwater and steam flow instruments are anticipated to be updated approximately every four minutes by the CROSSFLOW system. When the CROSSFLOW instrumentation is operating the secondary calorimetric power error contains an allowance (currently 31 days) for instrument drift. The four minute time interval during automatic operation is very small compared to the drift allowance. Therefore, the effect of instrument drift on the use of CROSSFLOW correction factors with the existing installed flow transmitters is considered and bounded.

Topical Report CEN-397(V)-P-A, Revision 01-P-A, "Improved Flow Measurement Accuracy Using Crossflow Ultrasonic Flow Measurements," January 2000.