March 14, 2000

Mr. W. Glenn Warren
BWR Owners Group Chairman
Southern Nuclear Company
42 Inverness Parkway
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Birmingham, AL 35201

SUBJECT: SAFETY EVALUATION OF GENERAL ELECTRIC NUCLEAR ENERGY TOPICAL REPORT B21-00658-01, "EXCESS FLOW CHECK VALVE TESTING RELAXATION" (TAC NOS. MA7884 AND M84809)

Dear Mr. Warren:

The Nuclear Regulatory Commission (NRC) staff has completed its review and evaluation of General Electric Nuclear Energy (GE) Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation," dated November 1998. This topical report was written by GE for the Boiling Water Reactor (BWR) Owners Group. It was submitted on April 12, 1999, by IES Utilities, Inc., the licensee for the Duane Arnold Energy Center (DAEC), as part of a plant-specific license amendment request. We approved the plant-specific license amendment on December 29, 1999. The topical report was reviewed for its generic applicability and the applicable safety evaluation (SE) is enclosed.

Based on the acceptability of the methods applied to estimate the release frequency, a relatively low release frequency estimate in conjunction with extremely low likelihood that this release could impact core damage frequency and negligible consequence of a release in the reactor building, we conclude that the increase in risk associated with DAEC's request for relaxation of excess flow check valve (EFCV) surveillance testing is sufficiently low and acceptable. With respect to other BWR's, we anticipate that similar conclusions can be drawn. Therefore, we consider the risk analysis portion of the topical report to be acceptable.

We also agree with the topical report that each plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. The topical report, however, lacks guidance for an individual plant to establish EFCV performance criteria and the basis which, we believe, is essential to ensure that a corrective action program provide meaningful feedback for appropriate corrective action. Therefore, we expect the BWR Owners Group to revise the topical report to either require each licensee to develop their EFCV performance criteria and the basis or develop an industry-wide performance criteria and the basis. As noted in the enclosed SE, the EFCV performance criteria should be based on sound reliability modeling that is consistent with generally expected performance of the EFCVs. We also note that such performance criteria and the basis, once developed, will be subject to staff review.

In conclusion, we find the topical report acceptable for referencing in relaxation of EFCV surveillance testing, subject to the conditions stated above. We also find the STS changes proposed by TSTF-334 to be acceptable, pending our acceptance of industry's development of EFCV performance criteria and the basis.

In accordance with procedures established in NUREG-0390, it is requested that the BWR Owners Group publish the accepted version of this report within three months of receipt of this letter. The accepted version should incorporate this letter and the appropriate evaluation between the title page and the abstract. The accepted version shall include an -A (designating accepted) following the report identification symbol.

Sincerely,

/***RA/***

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Project No. 691

Enclosure: Safety Evaluation

cc w/encl: See next page

In conclusion, we find the topical report acceptable for referencing in relaxation of EFCV surveillance testing, subject to the conditions stated above. We also find the STS changes proposed by TSTF-334 to be acceptable, pending our acceptance of industry's development of EFCV performance criteria and the basis.

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

BOILING WATER REACTOR OWNERS GROUP

GENERAL ELECTRIC NUCLEAR ENERGY TOPICAL REPORT B21-00658-01

"EXCESS FLOW CHECK VALVE TESTING RELAXATION"

1.0 INTRODUCTION

By letter dated April 12, 1999, as supplemented by letters dated October 5 and 8, 1999, Alliant Energy, the licensee for Duane Arnold Energy Center (DAEC), requested a license amendment which would allow relaxation of the frequency of surveillance testing of excess flow check valves (EFCVs) in reactor instrumentation lines. The proposed change was to relax the surveillance frequency by allowing a "representative sample" of EFCVs to be tested every 24 months, rather than having each EFCV tested every 24 months, as was previously required. The intent was to test approximately 20 percent of the EFCVs each 24 months such that each EFCV would be tested at least once every 10 years (nominal). The stated basis for the request was a high degree of reliability with the EFCVs and the low consequences of an EFCV failure. The analysis to support this conclusion was based on the General Electric Nuclear Energy Topical Report B21-00658-01, "Excess Flow Check Valve Testing Relaxation" (prepared for Boiling Water Reactor (BWR) Owners Group). The topical report was submitted as part of the Alliant Energy request for license amendment for DAEC.

We granted Alliant Energy's request with the issuance of Amendment No. 230 to Facility Operating License No. DPR-49 on December 29, 1999. As part of that action, we accepted the topical report insofar as it was applied to the DAEC case.

In this safety evaluation, we are reviewing the topical report for its generic applicability. By letter dated January 6, 2000, the BWR Owners Group provided additional information on the topical report, in the form of answers to our requests for additional information which we had made earlier during the DAEC review.

Furthermore, we are reviewing Standard Technical Specification Change Traveler TSTF-334, Revision 0, dated June 2, 1999, which was proposed by the industry Technical Specification Task Force. TSTF-334 implements the EFCV testing relaxation proposed by the topical report.

2.0 BACKGROUND

EFCVs in reactor instrumentation lines are used in BWR containments to limit the release of fluid from the reactor coolant system in the event of an instrument line break. Examples of EFCVs include reactor pressure vessel level/pressure instrument, main steam line flow instrument, recirculation pump suction pressure instrument, and reactor core isolation cooling steam line flow instrument. EFCVs are not required to close in response to a containment isolation signal and are not postulated to operate under post-LOCA conditions. The topical report states that EFCVs are not needed to mitigate the consequences of an accident because an instrument line break coincident with a design basis LOCA would be of a sufficiently low probability to be outside of the design basis.

BWR/4 Standard Technical Specification (STS) Surveillance Requirement (SR) 3.6.1.3.10 currently requires verification of the actuation (closing) capability of each reactor instrumentation line EFCV every [18] months. This is typical for the technical specifications (TS) at most operating BWR/4 plants. The proposed change is to relax the SR frequency by allowing a "representative sample" of EFCVs to be tested every [18] months. The intent is to test approximately 20 percent of the EFCVs every [18] months such that each EFCV will be tested at least once every 10 years (nominal). The proposed change is similar in principle to existing performance-based testing programs, such as inservice testing of snubbers and Option B of Appendix J to 10 CFR Part 50.

TSTF-334 includes a revised Basis for TS 3.6.1.3.10. The revised Basis states, in part:

> The representative sample consists of an approximately equal number of EFCVs, such that each EFCV is tested at least once every 10 years (nominal). In addition, the EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potentially common problem with a specific type or application of EFCV is detected at the earliest possible time.... The nominal 10 year interval is based on other performance-based testing programs, such as Inservice Testing (snubbers) and Option B to 10 CFR 50, Appendix J. Furthermore, any EFCV failures will be evaluated to determine if additional testing in that test interval is warranted to ensure overall reliability is maintained. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

3.0 EVALUATION

This evaluation has been divided into two parts, a systems review (3.1) and a risk and radiological review (3.2).

3.1 Systems Review

The topical report provides detailed information about EFCV surveillance testing at 12 BWR plants. Testing history indicates that there is a low failure rate in EFCV surveillance testing (see Section 3.2.1 below). Thus, EFCVs have generally been very reliable performers.

The dose consequences would be low if an EFCV failed to close if an instrument line broke during normal operation (see Section 3.2.2 below).

3.1.1 Request For Additional Information

The staff requested additional information from the BWR Owners Group regarding certain system aspects of their request, and they responded on January 6, 2000. The following three sections discuss these issues.

3.1.1.1 Test Interval Increase

The topical report compares this situation to Option B of Appendix J to 10 CFR Part 50. We revised Appendix J in 1995 by adding Option B, which provides a risk-informed, performance-based approach to leakage rate testing of containment isolation valves. We also developed Regulatory Guide 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, as a method acceptable to us for implementing Option B. This regulatory guide states that the Nuclear Energy Institute (NEI) guidance document NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to us for complying with Option B, with four exceptions which are described in the Regulatory Guide.

According to the NEI document, containment isolation valve test intervals may be increased to 5 years or 3 refueling outages if a valve has shown good performance (i.e., two consecutive successful tests), and, if certain other conditions are met, to as much as 10 years. However, the Regulatory Guide took exception to those provisions of the NEI document, stating that test intervals should not exceed 5 years. The Regulatory Guide explained that this was because of uncertainties (particularly unquantified leakage rates for test failures, repetitive/common mode failures, and aging effects) in historical containment isolation valve performance data, and because of the indeterminate time period of three refueling cycles and insufficient precision of programmatic controls described in the NEI document to address these uncertainties.

Topical Report B21-00658-01 states that the NEI document allows a 10-year test interval, and that Regulatory Guide 1.163 endorsed the NEI document, without mentioning our exception to 10 years. We asked the BWR Owners Group to justify their proposal for a 10-year testing interval.

Their response indicated that the topical report established its own basis for the testing relaxation, that being high reliability, low risk, and low radiological consequences. They have proposed a cyclic nominal interval for testing a representative sample of the EFCVs. The valves are of similar design, similar application, and similar service environment. Performance of the representative sample provides a strong indicator of the performance of the total population. The 10-year nominal interval solely limits the time between tests for any specific valve and provides additional assurance that all valves remain capable of performing their intended function. The BWR Owners Group considers the valve failure rate data listed in Table 4-1 of the topical report to be the primary basis for the performance-based interval. In addition, they assume that the off-site dose consequences of a failure to isolate have been evaluated and found to be acceptable, although each site adopting this change will need to confirm the applicability of this assumption.

Further, they did specifically address the reasons Regulatory Guide 1.163 had for not accepting a 10-year interval by providing the following information:

- Unquantified leakage rates for test failures are not applicable because the maximum leakage through an unisolated instrument line is quantified. The dose consequences of the failure to isolate are acceptable (see Section 3.2.2 below).

- Repetitive/common-mode failures are not applicable as evidenced by the low industry failure rate and more specifically by the Topical Report, Table 4-2, "EFCV Failure Rates by Manufacturer."

- Aging effects are not a concern. The industry data already provided does not indicate any increase in failure rate with time in service.

- Historical performance data associated with EFCVs has been provided and is considered adequate to justify the proposed interval.

- There is no indeterminate time period involved with this proposed change. Every fuel cycle, a representative sample will be tested.

We generally agree with the BWR Owners Group's assessment, except for the aging question which is addressed in Section 3.2.1 below. Regulatory Guide 1.163 had to consider all varieties of containment isolation valves, from a fraction of an inch to several feet in diameter, carrying liquid or gas in a wide range of temperatures and pressures. Different types of valves (gate, globe, check) made of various materials, by different manufacturers, and with varying safety significance, had to be accounted for. On the other hand, EFCVs in reactor instrumentation lines are a very specific, narrow class of valves. Their history and performance are well-documented. Based on their historically high reliability and their low risk significance and radiological consequences should they fail (as discussed in Section 3.2, below), we accept the proposal that the magnitude of a test interval extension may be as great as 10 years.

3.1.1.2 Failure Feedback Mechanism

In our second question, we pointed out that, under Appendix J, Option B, testing programs, a valve that fails a test after having been put on an extended test interval must return to its original interval until it once again shows good performance (i.e., passes two consecutive tests). Risk-informed inservice testing Regulatory Guide 1.175, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Inservice Testing," also specifies the need for a failure feedback mechanism. Topical Report B21-00658-01 has no specific failure feedback mechanism, although it does state that each plant's corrective action programs must evaluate equipment failures and establish appropriate corrective actions.

The BWR Owners Group replied that each licensee who adopts the relaxed surveillance intervals recommended by the topical report should ensure that an appropriate feedback mechanism to respond to failure trends is in place, and that TSTF-334 includes this commitment (see the proposed Basis for TSTF-334, quoted in Section 2.0, above).

Considering the historically high reliability of the EFCVs and their low risk significance and radiological consequences should they fail, we find that it is not necessary for the topical report to provide a specific failure feedback mechanism. However, see Section 3.2.1 below for an additional discussion of the corrective action program and EFCV performance criteria.

3.1.1.3 Technical Specification Level of Detail

In TSTF-334, the proposed TS says "a representative sample" of EFCVs will be tested every [18] months. The "representative sample" is not defined in the TS itself. The proposed Basis says that a licensee will test 20 percent of the valves each refueling outage and thus test all of them in a 10-year period. We asked the BWR Owners Group to justify having the specific requirements in the Bases, rather than in the proposed TS.

They replied that the term "representative sample," with an accompanying explanation in the TS Bases, is identical to current usage in the STS, NUREG-1433, Revision 1. Specifically, NUREG-1433 uses the term "representative" in TS SR 3.8.6.3, in reference to battery cell testing, and "representative sample" in SR 3.1.4.2 for verification of control rod scram times. Therefore, the application of a "representative sample" for the EFCV testing SR, with its accompanying definition in the Bases, is consistent with the STS usage.

They also provided additional examples and explanations that supported their proposed TS as being consistent with current STS practices.

Therefore, we find the proposed TS and Bases wording to be acceptable.

3.2 Risk and Radiological Review

Below is our review of the risk and radiological analysis associated with this request. Specifically, we evaluated: (1) the estimate of the steam release frequency (into the reactor building) due to a break in an instrument line concurrent with an EFCV failure to close and (2) the assessment of the radiological consequence of such release.

The instrument lines at DAEC include a 1/4 inch flow restriction orifice upstream of the EFCVs to limit reactor water leakage in the event of a rupture. As discussed below, in Section 3.2.2, previous evaluation of such an instrument line rupture in DAEC Updated Final Safety Analysis Report (UFSAR) 1.8.11, for which the EFCVs are designed to mitigate, do not credit the isolation of the line by the EFCVs. Thus, a failure of an EFCV is bounded by the previous evaluation of an instrument line rupture. This analysis also showed that the resulting offsite doses would be well below regulatory limits. Further discussion on radiological impact is provided in Section 3.2.2 of this safety evaluation.

The operational impact of an EFCV that is connected to the reactor pressure vessel (RPV) boundary failing to close is based on the environmental effects of a steam release in the vicinity of the instrument racks. The environmental impact of the failure of instrument lines connected to the RPV pressure boundary is the released steam into the reactor building. However, the topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of reactor building ventilation systems and that the integrity and functional performance of secondary containment following instrument line break would be met.

The separation of equipment in the reactor building is also expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement. Nevertheless, the presence of an unisolated steam leak into the reactor building requires the licensee to shut down the reactor and depressurize to allow access to manually isolate the line.

The BWR Owners Group estimation of the steam release frequency caused by an instrument line break concurrent with an EFCV failing to close is reviewed in Section 3.2.1 of this report. The assessment of the radiological consequences of such release is reviewed in Section 3.2.2.

3.2.1 Estimation of Release Frequency

In estimating the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency and (2) the probability of EFCV failing to close. The BWR Owners Group assumed a single instrument line break frequency of 3.52E-05/year. This estimate was based on the EPRI Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants", dated July 1992. This frequency corresponded to pipe sizes between ½ inch to 2 inches in diameter and the BWR Owners Group considered these pipe sizes to represent the subject instrument line piping. Thus, for DAEC, the product of this single instrument line break frequency and the total number of instrument lines at DAEC, 94, resulted in the total plant instrument line break frequency estimate of 3.31E-3/year.

Since the above single instrument line break frequency represents recent data, we consider its application to estimate the plant instrument line break frequency to be acceptable. We note, however, that the total plant instrument line break frequency for a plant depends on the total number of instrument lines present at the plant.

The probability of EFCV failing to close (or EFCV unavailability) was estimated using the formula:

$$\bar{A} = \lambda * \theta / 2$$

Where:
- $\bar{A}$ is the EFCV unavailability
- λ is the EFCV failure rate per year
- θ is the EFCV surveillance test interval in years

The EFCV failure frequency, λ, was estimated using the formula:

$$\lambda_u = \theta^2_{\theta;2r+2} / 2T$$

Where :
- λ_u is the upper limit failure rate per year
- T is the operating time in years
- r is the number of failures
- $\theta^2_{\theta;2r+2}$ is the value taken from the chi-square distribution tables which corresponds to 2r+2 degrees of freedom at $\theta = 0.05$ (0.95 confidence level)

The topical report determined an upper limit EFCV failure rate based on 11 observed failures in about 12424.5 years of service for 12 BWR plants in the U.S. (Note: 12424.5 years was determined by multiplying the number of tested EFCVs with the time period during which the number of occurring failures was reported). For eleven observed EFCV failures, the EFCV upper limit failure rate, λ_u, was estimated to be about 1.5E-3/year.

It is noted that the formula for estimating the EFCV failure rate, λ_u, assumes that this failure rate is constant over time. Therefore, to account for the possibility that the failure rate for EFCV may change over time, potentially due to age-related factors, the EFCV failure rate was assumed to change by five fold in the report's analysis. We consider the use of this method to be acceptable.

In addition to accounting for a potential age-related degradation in the EFCV failure rate estimate, the topical report stated that each plant's corrective action programs must evaluate equipment failures and establish appropriate corrective actions. We consider this requirement to be prudent and necessary. However, to ensure that such a program can provide meaningful feedback for appropriate corrective actions, we believe that the topical report should require each licensee to develop their EFCV performance criteria and the basis.

For 55 EFCV failures (5 times the actual number of EFCV failures observed for 12 BWR plants), degrees of freedom (2r +2, where r is the number of failures) is 112. Chi-squared values, $\theta^2_{\theta;2r+2}$, are not typically provided for degree of freedom values above 30 because a chi-squared distribution with degrees of freedom over 30 approximates the standard normal distribution. In such case θ^2 is approximated by:

$$\boldsymbol{\theta^2 = \tfrac{1}{2} (Z + (2n-1)^{1/2})^2}$$

Where: - Z is the corresponding standard deviation (or a z-score) for θ-point of the standard normal distribution
- n is the degrees of freedom

Thus, for a 0.95 confidence level (θ = 0.05), Z is 1.645. And,

$$\theta^2 = \tfrac{1}{2} (1.645 + (2*112-1)^{1/2})^2 = 137.42$$

Therefore, EFCV upper limit failure frequency was then calculated to be:

$$\lambda_u = \theta^2 / 2T = 137.42 / (2 * 12424.5 \text{ years}) = 5.53\text{E-3 failures per year}$$

The release frequency was then calculated by the formula:

$$\begin{aligned} \mathbf{RF} &= \mathbf{I * \bar{A}} \\ &= \mathbf{I * \lambda_u * \theta / 2} \end{aligned}$$

Where:
- RF is the release frequency
- I is the instrument line failure frequency (per year)
- Ā is the EFCV unavailability (calculated by λ * θ / 2)
- θ is the surveillance interval in years

Using the surveillance interval for 2 years (current practice), the instrument line break frequency of 3.31E-3/year at DAEC, and total plant EFCV failure frequency of 5.53E-3/year, the release frequency was estimated to be 1.8E-5/year. For a surveillance interval of 10 years, the release frequency was estimated to be about 9.1E-5/year, which depicts an increase of about 7.3E-5/year from that of the 2-year surveillance test interval. It represents the increase in the total plant release frequency for a random break of any of the 94 instrument lines in DAEC and a concurrent failure of the line's EFCV to close to isolate the break.

We do not consider this estimated increase in release frequency, 7.3E-5/year, to be significant. This conclusion is based on the point that this frequency is lower than the DAEC large-break loss of coolant accident (LOCA) frequency of 3E-4/year which has the potential to lead to a core damage accident whereas the instrument line break concurrent with EFCV failing to close does not.

In addition, we consider the above method for assessing the impact of EFCV surveillance test interval increase to 10 years (along with an assumed five-fold increase in the EFCV failure rate) to be acceptable. We note that the use of observed industry data for instrument line break and EFCV failures is sound for DAEC's case. However, for a plant whose instrument line break frequency and/or EFCV failure rate exceed that of the industry average, the plant-specific data should be applied in the estimation of the release frequency. We also recognize that the method of estimating the EFCV unavailability is consistent with industry practice and that accounting for a potentially unknown change in the valve's failure rate is prudent.

3.2.2 Radiological Consequences

The lead plant, DAEC, noted that they previously evaluated the radiological consequences of an unisolable rupture of such an instrument line in response to Regulatory Guide 1.11, as documented in DAEC UFSAR 1.8.11. This evaluation assumed a continuous discharge of reactor water through an instrument line with a 1/4-inch orifice for the duration of the detection and cooldown sequence. The assumptions for the accident evaluation do not change as a result of the proposed TS change, and the evaluation in DAEC UFSAR 1.8.11 remains acceptable. Therefore, we find acceptable the licensee's determination that the proposed amendment will not involve a significant increase in the consequences of an accident previously evaluated.

The topical report also maintained that radiological consequences from reactor coolant pressure boundary instrument line breaks have been evaluated at most plants to show compliance with Regulatory Guide 1.11 and are documented in some UFSARs. A typical GE radiological evaluation of the instrument line break with and without a 1/4 inch orifice installed has been conducted using a GE methodology which has been accepted by us in BWR FSAR submittals. The results of the evaluation indicated that even without a 1/4 inch orifice installed, the resulting thyroid dose at the site boundary is about 5 percent of the regulatory limit. The report concluded, therefore, that the radiological consequence of EFCVs failing to function upon demand is sufficiently low to be considered insignificant. The report further stated that specific analyses are needed to confirm this conclusion at each plant, but that similar results would be expected.

4.0 CONCLUSION

As demonstrated in the topical report, the impact of an increase in EFCV surveillance test intervals to 10 years along with an assumed five-fold increase in the EFCV failure rate on the likelihood of a release inside the reactor building was shown to result in a release frequency of about 9.1E-5/year for DAEC. This represents an increase of about 7.3E-5/year from the current release frequency estimate (for 2-year surveillance test interval) of about 1.8E-5/year. We consider this estimate for DAEC to be sufficiently low, especially since the consequence of such an accident is not expected to lead to a core damage. For some BWR plants, the estimated release frequency may be higher than DAEC's estimate when plant-specific instrument line break frequency and/or EFCV failure rate (that are higher than the industry average) are used in the calculation. However, based on the reported individual plant EFCV failure data, we do not anticipate significant deviation from the estimate derived for DAEC.

We also agree with the topical report that the consequences of steam release from the depicted events is not significant, as it was supported by a previous analysis. Based on the acceptability of the methods applied to estimate the release frequency, a relatively low release frequency estimate in conjunction with unlikely impact on core damage and negligible consequence of a release in the reactor building, we conclude that the increase in risk associated with DAEC's request for relaxation of EFCV surveillance testing is sufficiently low and acceptable. With respect to other BWRs, we anticipate that similar conclusions can be drawn. Therefore, we consider the risk analysis portion of the topical report to be acceptable.

We also agree with the topical report that each plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. The topical report, however, lacks guidance for an individual plant to establish EFCV performance criteria and the basis which, we believe, is essential to ensure that a corrective action program can provide meaningful feedback for appropriate corrective action. Therefore, we expect the BWR Owners Group to revise the topical report to either require each licensee to develop their EFCV performance criteria and the basis or develop an industry-wide performance criteria and the basis. The EFCV performance criteria should be based on sound reliability modeling that is consistent with generally expected performance of the EFCVs. We also note that such performance criteria and the basis, once developed, will be subject to staff review.

In conclusion, we find the topical report acceptable for referencing in relaxation of EFCV surveillance testing, subject to the conditions stated above. We also find the STS changes proposed by TSTF-334 to be acceptable, pending our acceptance of industry's development of EFCV performance criteria and the basis. We also note, as stated in TSTF-334, that some plants may require an inservice testing program relief request pursuant to 10 CFR 50.55a in order to implement these TS changes.

Principal Contributor: S. Lee

Date: March 14, 2000