January 9, 2008

Mr. Gordon Bischoff, Manager
Owners Group Program Management Office
Westinghouse Electric Company
P.O. Box 355
Pittsburgh, PA 15230-0355

SUBJECT: FINAL SAFETY EVALUATION FOR PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG) TOPICAL REPORT (TR) BAW-10179(P), REVISION 7, “SAFETY CRITERIA AND METHODOLOGY FOR ACCEPTABLE CYCLE RELOAD ANALYSES” (TAC NO. MC9607)

Dear Mr. Bischoff:

By letter dated December 23, 2005, the Babcock & Wilcox Owners Group, now known as the PWROG, submitted TR BAW-10179(P), Revision 7, “Safety Criteria and Methodology for Acceptable Cycle Reload Analyses” to the U.S. Nuclear Regulatory Commission (NRC) staff for review. By letter dated December 14, 2007, an NRC draft safety evaluation (SE) regarding our approval of TR BAW-10179(P), Revision 7, was provided for your review and comments. By letter dated December 20, 2007, the PWROG provided text and typographical clarification to the draft SE. The editorial corrections are reflected in the enclosed final SE.

The NRC staff has found that TR BAW-10179(P), Revision 7, is acceptable for referencing in licensing applications for Babcock & Wilcox designed pressurized water reactors to the extent specified and under the limitations delineated in the TR and in the final SE. The final SE defines the basis for our acceptance of the TR.

Our acceptance applies only to material provided in the subject TR. We do not intend to repeat our review of the acceptable material described in the TR. When the TR appears as a reference in license applications, our review will ensure that the material presented applies to the specific plant involved. License amendment requests that deviate from this TR will be subject to a plant-specific review in accordance with applicable review standards.

In accordance with the guidance provided on the NRC website, we request that PWROG publish accepted proprietary and non-proprietary versions of this TR within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed final SE after the title page. Also, they must contain historical review information, including NRC requests for additional information and your responses. The accepted versions shall include an "-A" (designating accepted) following the TR identification symbol.

If future changes to the NRC's regulatory requirements affect the acceptability of this TR, the PWROG and/or licensees referencing it will be expected to revise the TR appropriately, or justify its continued applicability for subsequent referencing.

Sincerely,

/RA/

Ho K. Nieh, Deputy Director
Division of Policy and Rulemaking
Office of Nuclear Reactor Regulation

Project No. 694

Enclosure: Final SE

cc w/encl:
Mr. James A. Gresham, Manager
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FINAL SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TOPICAL REPORT (TR) BAW-10179(P), REVISION 7

"SAFETY CRITERIA AND METHODOLOGY FOR ACCEPTABLE

CYCLE RELOAD ANALYSES"

PRESSURIZED WATER REACTOR OWNERS GROUP (PWROG)

PROJECT NO. 694

1.0 INTRODUCTION AND BACKGROUND

By letter dated December 23, 2005, the Babcock & Wilcox Owners Group, now known as the PWROG, submitted TR BAW-10179(P), Revision 7, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" to the U.S. Nuclear Regulatory Commission (NRC) staff for review. As designer and fuel fabricator, AREVA NP, Inc. (AREVA) prepares reload safety evaluations (SEs) for a number of Babcock & Wilcox (B&W)-designed nuclear power plants with 177 fuel assemblies. The methodology for performing reload design evaluations is presented in TR BAW-10179P-A, "Safety Criteria and Methodology for Acceptable Cycle Reload Analyses" (Reference 1). The utility owners for whom AREVA performs reload SEs, reference TR BAW-10179P-A in the Administrative Controls sections of the plant Technical Specifications (TSs). The TSs identify TR BAW-10179P-A as the NRC-approved methodology for determining the limits contained in the Core Operating Limits Report (COLR). The TSs also state that the latest approved revision of BAW-10179P-A shall be specified in the COLR. Since the original approval of TR BAW-10179P-A (Reference 1), six revisions to the report have been issued in the form of multiple appendices to incorporate additional NRC-approved codes and methods.

The purpose of Revision 7 is to:

- Incorporate the appendices from Revisions 1 through 6 in to the main body of the report,
- Update the methodology to incorporate the NRC-approved design code COPERNIC,
- Update the methodology to incorporate the NRC-approved statistical fuel assembly hold-down methodology,
- Provide a summary of the NRC-approved modified zero power physics testing program,
- Add clarification where needed and remove unnecessary information,

ENCLOSURE

- Provide generic guidelines on the use of limited scope high burnup lead test assemblies and satisfy the requirement to incorporate WCAP-15604-NP, Revision 2-A ("Limited Scope High Burnup Lead Test Assemblies") explicitly into the licensee's TSs by reference in TR BAW-10179P-A.

TR BAW-10179(P), Revision 7, describes the entire spectrum of methodologies applicable to reload fuel currently supplied by AREVA for the B&W-designed nuclear power plants with 177 fuel assemblies.

2.0 REGULATORY EVALUATION

Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix A, "General Design Criteria for Nuclear Power Plants," Criterion 10, "Reactor design," requires that: The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits [(SAFDLs)] are not exceeded during any condition of normal operation, including the effects of anticipated occurrences.

Criterion 11, "Reactor inherent protection," requires that: The reactor core and associated coolant systems shall be designed so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tend to compensate for a rapid increase in reactivity.

Criterion 12, "Suppression of reactor power oscillations," requires that: The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding the SAFDLs are not possible, or can be reliably and readily detected and suppressed.

Additionally, 10 CFR Part 50, Section 34, "Contents of applications; technical information," requires that Safety Analysis Reports be submitted that analyze the design and performance of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents. As part of the core reload design process, licensees (or vendors) perform reload safety evaluations to ensure that their safety analyses remain bounding for the design cycle. To confirm that the analyses remain bounding, licensees confirm those key inputs to the safety analyses are conservative with respect to the current design cycle. If key safety analysis parameters are not bounded, a reanalysis or a reevaluation of the affected transients or accidents is performed to ensure that the applicable acceptance criteria are satisfied.

In Generic Letter 88-16, "Removal of Cycle-Specific Parameter Limits from Technical Specifications," the NRC allowed for the removal of cycle-dependent variables from the TSs, provided the values of these variables are determined with an NRC-approved methodology and are included in a COLR. Additionally, the NRC agreed (Reference 2) that this approach can be extended to the cycle-dependent protective and maximum allowable setpoint limits. Therefore, licensees with B&W designed nuclear power plants now use TR BAW-10179-P-A as the approved methodology to ensure compliance with Criteria 10, 11, and 12 of 10 CFR Part 50, Appendix A.

In its review of the proposed modifications in Revision 7 of TR BAW-10179-P, the NRC staff ensured that the changes continued to meet the above regulatory requirements.

TECHNICAL EVALUATION OF THE REVISED REPORT SECTIONS

TR BAW-10179(P), Revision 7, consists of 11 sections. Section 1, The Introduction, reviews the original objectives and the evolution of TR BAW-10179P-A. Section 1 also gives the motivation for the submission of TR BAW-10179(P), Revision 7, for review and approval by the NRC.

Section 2, Design Considerations, describes and compares, as a function of event frequency, the categories of events as defined by NRC in Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, and those used by AREVA in the context of TR BAW-10179(P) and defined by American National Standards Institute/American Nuclear Society (ANSI/ANS) - 57.5. The NRC categorizes plant operation into three conditions: normal operation, anticipated operational occurrences, and accidents. AREVA categorizes plant operation into four conditions: condition I through IV. AREVA assures compliance with the NRC regulations in Section 2.0, Regulatory Evaluation, of this Draft SE by requiring the limiting Condition III transient to meet the acceptance criteria for Condition II events.

Section 3, Reference Fuel Description for Mark-B Design, describes the fuel pellet, fuel rod, and fuel assembly of the Mark-B fuel designs incorporating Zircaloy-4 and M5 material. Section 4, Mechanical Safety and Design Criteria, describes the design criteria and the analysis method for the fuel assembly design and the fuel rod design. These criteria and methods will ensure that under all operating conditions the maximum credible damage will not degrade the design below those capabilities assumed in the safety analysis.

Section 5, Nuclear Design, describes fuel cycle design and analysis, maneuvering analysis, and nuclear analysis. These analyses demonstrate that the maximum calculated steady-state fuel rod relative power density for the cycle shall be less than that required to meet the safety criteria, that the discharge burnup shall be less than the applicable limit for that fuel design, and that the design shall be capable of passing subsequent safety analysis checks.

Section 6, Core Thermal Hydraulics, describes the analyses performed to assure that two acceptance criteria are met. The departure from nucleate boiling (DNB) ratio safety criterion states that there shall be at least a 95 percent probability at a 95 percent confidence level that the hot fuel rod in the core does not experience DNB during normal operation or events of moderate frequency. The second criterion states that the worst case hydraulic loads should not exceed the hold-down capability of the fuel assembly during normal operation.

Section 7, Reactor Protection System Setpoints, discusses the trip setpoint calculations for the analyses: high reactor coolant system pressure trip; low reactor coolant system pressure trip; power/imbalance/flow trip; high flux trip; high reactor coolant outlet temperature trip; variable low RC pressure trip; high reactor building pressure trip; power/pump monitor trip. Section 8, Non-LOCA [loss-of-coolant accident] Accident Evaluation, discusses the assumptions used in the Safety Analysis Report (SAR) with regard to non-LOCA analyses that must be shown to bound the corresponding cycle-specific parameters in order to avoid a reevaluation of the accident for each fuel cycle. Thus, the predicted parameter values of a reload fuel cycle must be shown to lie within the current licensing basis of the plant. This section discusses how each of the key parameters is evaluated for each fuel cycle against the licensing analysis for each plant.

Section 9, Loss-of Coolant Accident Evaluation, describes the use of LOCA limits as input to the reload SE. That is, given that the most likely change in a fuel reload that could impact the LOCA analysis is a change in the peaking, this change is controlled within a set of maximum allowed LOCA kW/ft limits, defined to represent initial condition requirements for the LOCA analysis. Section 10, Limited Scope High Burnup Lead Test Assemblies [LTA], provides generic guidelines on the use of limited scope high burnup LTAs and satisfies the requirement to incorporate TR WCAP-15604-NP, Revision 2-A, "Limited Scope High Burnup Lead Test Assemblies," explicitly into the licensee's TSs by virtue of being referenced in TR BAW-10179(P), Revision 7. Section 11 provides a list of references.

A significant difference between Revision 7 of TR BAW-10179(P), under review, and previously approved revisions of TR BAW-10179P-A is the incorporation, by reference, of the information in Appendices A through X from the previously approved revisions into the text of Revision 7. Based on these previous approvals, the NRC staff considers the information presented in these appendices and the associated TRs acceptable.

3.1 Summary of the Key Content of TR BAW-10179(P)-A, Revision 1 through 6 Appendices

Appendix A: Lists the relevant reports that received NRC approval subsequent to the submittal of TR BAW-10179P-A and are incorporated in TR BAW-10179(P), Revision 7. These reports are described in Appendix B through X.

Appendix B: "Stainless Steel Replacement Rod Methodology," BAW-2149-A. The approved TR of this appendix addresses the nuclear, thermal-hydraulic, and mechanical aspects of the design that are affected by in-field repair of irradiated assemblies. In particular, it justifies the use of replacement rods without imposing unnecessary power peaking restrictions on the repaired fuel assembly.

Appendix C: "LYNXT Thermal-Hydraulics Code," BAW-10156-A, Revision 1. This TR introduces a supplemental solution technique to LYNXT that provides additional capabilities as an alternative to the original solution methods.

Appendix D: "Statistical Core Design [SCD] for B&W-Designed 177FA Plants," BAW-10187P-A. This TR introduces a more realistic assessment of core DNB protection, called SCD, to treat the core state and bundle uncertainties. The method reduces some of the undue conservatism of the deterministic approach, while still allowing for the traditional compounding of variables not amenable to statistical treatment.

Appendix E: "GDTACO: Gadolinia Fuel Rod Thermal Analysis Code," BAW-10184P-A. The GDTACO code is a modification of the approved TACO3 code that incorporates the physical material properties of gadolinium oxide for calculating the thermal performance of urania-gadolinia fuel rods.

Appendix F: “Fuel Rod Gas Pressure Criterion (FRGPC),” BAW-10183P-A. This new criterion addresses the increasing fuel rod internal pressure due to an extension of the burnup of the nuclear fuel.

Note: The following appendices were added to Appendix A during subsequent revisions of TR BAW-10179P up through Revision 6.

Appendix G: NRC approval for the use of the Nodal Expansion Method Optimized (NEMO) code uncertainty in TACO3 fuel performance calculations.

Appendix H: “Program to Determine In-Reactor Performance of BWFC Fuel Cladding Creep Collapse,” BAW-10084P-A, Revision 3. BAW-10084P-A, Revision 3, introduces an improved methodology that provides more realistic estimates of creep collapse timing and is used to demonstrate that clad flattening will not occur.

Appendix I: “Extended Burnup Evaluation,” BAW-10186P-A. BAW-10186P-A introduces improved evaluation methods to demonstrate acceptable fuel performance at higher burnup levels.

Appendix J: Letter from NRC to Framatome Cogema Fuels, “Concerning the Extension to the TACO3 Burnup Limit to 62 GWd/mtU.” With this letter, the NRC approves an increase in the TACO3 burnup limit from 60 to 62 GWD/mtU.

Appendix K: Letter from NRC to Framatome Cogema Fuels, “Accepting Revised Measurement Uncertainty for Control Rod Worth Calculations.” With this letter, the NRC approves a 5 percent uncertainty on total rod worth values computed with NEMO.

Appendix L: “The BWU Critical Heat Flux [CHF] Correlations,” BAW-10199P-A. BAW-10199P-A introduces a universal local conditions CHF correlation for extended applications to middle or low flow regimes.

Appendix M: Letter from NRC to B&WOG,“Safety Evaluation of the Babcock & Wilcox Owners Group Submittal Relating to Assumptions in the B&W ECCS [Emergency Core Cooling System] Analysis.” This appendix replaces the reference to BAW-1781P in Section 4.1.5.1 of BAW-10179P-A as it relates to horizontal faulted LOCA-seismic structural evaluation.

Appendix N: “BWNT Loss of Coolant Accident Evaluation Model for Once-Through Steam Generator Plants,” BAW-10192P-A.

Appendix O: “NEMO-K A Kinetics Solution in NEMO,” BAW-10221P-A. A methodology for the analysis of three-dimensional time-dependent solutions for neutronics, fuel temperature, and coolant properties.

Appendix P: “Mark-C Fuel Assembly LOCA-Seismic Analyses,” BAW-10133P-A, Revision 1, Addendum 1 and Addendum 2. Addendum 1 provides changes to the modeling of the spacer grids and the modeling of the hydrodynamic coupling. Addendum 2 provides the justification of higher damping values in fuel assembly seismic and LOCA models.

Appendix Q: "The BWU Critical Heat Flux Correlations," BAW-10199P-A, Addendum 1. This appendix justifies the application of the BWU-Z CHF correlation to fuel assemblies with Mark-B11 spacer grid design.

Appendix R: "Evaluation of Advanced Cladding and Structural Material M5 in PWR [Pressurized Water Reactor] Reactor Fuel," BAW-10227P-A. This appendix justifies the use of alloy M5 to replace Zircaloy-4 in the construction of fuel assembly components.

Appendix S: "SCIENCE," BAW-10228P-A. This report describes a nuclear code package and its application to fuel cycle design and licensing.

Appendix T: "Mark-B11 Fuel Assembly Design Report," BAW-10229P-A. This report provides the licensing bases for the Mark-B11 fuel assembly design.

Appendix U: "RELAP5/MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis," BAW-10164P-A, Revision 4. This report describes the RELAP5/MOD2-B&W computer code utilized in the BWNT LOCA EM TR BAW-10192P-A discussed in Appendix N.

Appendix V: "BHTP DNB Correlation Applied with LYNXT," BAW-10241P-A. This report provides the technical justification for using the HTP correlation in the LYNXT thermal-hydraulic code.

Appendix W: "BEACH - Best Estimate Analysis Core Heat Transfer, A Computer Program for Reflood Heat Transfer during LOCA," BAW-10166P-A, Revision 5. This report describes the BEACH computer code utilized in the BWNT LOCA Evaluation Model for Once-Through Steam Generator Plants, TR BAW-10192P-A, that is discussed in Appendix N.

Appendix X: "BHTP DNB Correlation Applied with LYNXT," BAW-10241P-A, Revision 1. The limitations of the BHTP CHF correlation to specific local coolant condition ranges are expanded in BAW-10241P-A, Revision 1, through the introduction of additional CHF data supporting the extensions.

3.2 Location of the Appendices in the text of TR BAW-10179(P), Revision 7

TR BAW-10179(P), Revision 7, incorporated the Appendices from Revisions 1 through 6 of the TR and incorporated them into the body of the text by reference.

3.2.1 Section 4: Mechanical Safety and Design Criteria

4.1.7.1 Analysis Criteria: Appendix I as Reference 9

4.1.7.2 Analysis Methods: Appendix I as Reference 9

4.1.9.2 Analysis Methods: Appendix M as Reference 16 and Appendix P as Reference 12

4.1.11.3 Evaluation of Advanced Cladding and Structural Material (M5): Appendix R as Reference 18

4.1.12 Extended Burnup: Appendix I as Reference 9

4.1.13 Stainless Steel Replacement Rod Methodology: Appendix B as Reference 23

4.2 Fuel Rod Design: Appendix R as Reference 18

4.2.3.1 Analysis Criteria: Appendix I as Reference 9

4.2.3.2 Analysis Methods: Appendix I as Reference 9

4.2.3.3 Lead Corrosion Assemblies: Appendix I as Reference 9

4.2.4.2 Analysis Methods: Appendix H as Reference 26

4.2.5.1 Analysis Criteria: Appendix R as Reference 18

4.2.8.2 Analysis Methods: Appendix E as Reference 28, Appendix F as Reference 31, Appendix G as Reference 30 and Appendix J as Reference 29

4.2.9 Fuel Temperature (Centerline Fuel Melt) Limit: Appendix J as Reference 29

4.2.9.2 Analysis Methods: Appendix E as Reference 28

3.2.2 <u>Section 5: Nuclear Design</u>

5.1 Nuclear Design Codes: Appendix O as Reference 34

5.2.1.2 Final Fuel Cycle Design: Appendix B as Reference 23

5.2.1.1 Acceptance Criteria: Appendix I as Reference 9

5.3.1.3 Axial Power Imbalance Protective Limits: Appendix B as Reference 23

5.4 Nuclear Parameters For Safety Analysis: Appendix B as Reference 23

5.4.2.1.2 Analysis Methods: Appendix K as Reference 41

5.4.2.2.2 Analysis Methods: Appendix K as Reference 41

3.2.3 <u>Section 6: Core Thermal Hydraulics</u>

6.1 Design Criteria:

- Appendix C as Reference 50
- Appendix L as Reference 46
- Appendix Q as Reference 51
- Appendix V as Reference 4
- Appendix X as Reference 4

6.2.1.3 LYNXT: Appendix C as Reference 50

6.2.9 Statistical Core Design: Appendix D as Reference 62

6.3.2 Core Pressure Drop: Appendix C as Reference 50

6.8 Stainless Steel Replacement Rod Methodology: Appendix B as Reference 23

3.2.4 <u>Loss-of-Coolant Accident Evaluation</u>

9.2 Analysis Methods: Appendix N as Reference 69

9.2.2 RELAP5 EM: Appendix N as Reference 69, Appendix R as Reference 18, Appendix U as Reference 75 and Appendix W as Reference 77

9.2.3 Steady-State Fuel data Input to LOCA EMs: Appendix E as Reference 28 and Appendix I as Reference 9

9.3 Generic LOCA Evaluations: Appendix B as Reference 23 and Appendix R as Reference 18

9.4 Application Of Generic Evaluations To Cycle-Specific Plant Conditions: Appendix B as Reference 23

3.3 Incorporation of Fuel Rod Design Code COPERNIC as Reference 25

4.2.2.2 Plenum Space Analysis Method: As an alternative to the code TACO3 (Reference 24) in computing the stack hot length and cladding expansion for a high power history.

4.2.3.2 Corrosion Analysis Method: Corrosion predictions for M5 clad fuel

4.2.4.2 Creep Quality Analysis Method: Fuel rod simulations for obtaining temperatures and pressures as inputs to the CROV code

4.2.6.2 Strain Analysis Method: As an alternative to the TACO3 code for prediction of fuel rod cladding strain

4.2.8.2 Fuel Rod Pressure Analysis Method: As an alternative to the codes TACO3 and GDTACO for fuel design reference analysis

4.2.9.2 Fuel Temperature (Centerline Fuel Melt) Limit: As an alternative to the codes TACO3 and GDTACO for fuel design reference analysis

3.4 Modified Zero Power Physics Testing Program

Reload startup physics testing is performed following refueling outages to verify that the core is operating as designed. The standard startup physics testing scope for B&W-designed plants complies with ANSI/ANS 19.6.1. The inclusion of Reference 81, BAW-10242-A, “ZPPT Modifications for B&W Designed Reactors,” justifies three NRC-approved modifications to ANSI/ANS 19.6.1 for the ZPPT program for B&W-designed reactors.

3.5 Section 10: Limited Scope High Burnup Lead Test Assemblies

This section summarizes the requirements for a limited scope, high burnup LTA program. Sections 1 through 9 of TR BAW-10179(P), Revision 7, describe the methods and models used to evaluate core and fuel assembly performance up to the current burnup limits. For purposes of evaluating the performance of the LTAs beyond current burnups limits, as part of the safety analysis, an assessment must be made of the models that have been reviewed and approved by the NRC staff. The analytical models used to evaluate the performance of the LTAs beyond the current burnup limits may need to be modified versions of the models reviewed and approved by the NRC staff. The revised models would be used only for the limited scope high burnup LTAs and not for any other assemblies in the core. The justification of the model revisions are to be documented and made available for NRC review in accordance with the 10 CFR 50.59 criteria.

4.0 CONCLUSION

The NRC staff has reviewed TR BAW-10179(P), Revision 7. The TR incorporates into the original NRC-approved report (BAW-10179P-A, Revision 0) six revisions in the form of multiple appendices that incorporated additional NRC-approved codes and methods. In addition, incorporated are the NRC-approved mechanical design code COPERNIC, and a NRC-approved modified zero power physics testing program for the B&W-designed nuclear power plants with 177 fuel assemblies.

The NRC staff concurs that the purpose of Revision 7 has been met through:

1. The incorporation of the appendices from Revisions 1 through 6 into the main body of the TR.

2. Updating the methodology to incorporate the NRC-approved design code, COPERNIC.

3. Updating the methodology to incorporate the NRC-approved statistical fuel assembly hold-down methodology.

4. A summary of the modified zero power physics testing program.

5. The clarification where needed and the removal of unnecessary information.

6. Generic guidelines on the use of limited scope high burnup LTAs and incorporation of WCAP-15604-NP, Revision 2 A ("Limited Scope High Burnup Lead Test Assemblies") explicitly into the licensee's TSs by virtue of it being referenced in TR BAW-10179P-A.

On the basis of this review, the NRC staff concludes that the proposed modifications to TR BAW-10179-P, Revision 7 will continue to meet the applicable regulatory requirements listed in Section 2.0 of this SE. Therefore, the NRC staff approves the safety criteria and methodology for acceptable cycle reload analysis as documented in TR BAW-10179(P), Revision 7.

5.0 REFERENCES

1. Ashok C. Thadani (NRC) letter to Joseph D. McCarthy, B&WOG, Subject: Acceptance for Referencing of Licensing Topical Report BAW-10179-P, "Safety Criteria and Methodology for acceptable Cycle Reload Analyses," March 16, 1993.

2. Memorandum for Jose A. Calvo from David C. Fisher, Subject: Minutes of Meeting with the B&W Owners Group (B&WOG) on the Technical Basis and Scope of the BWOG Core Operating Limits Report, July 19, 1989.

Principal Contributor: Yuri Orechwa

Date: January 9, 2008