December 5, 2007

Mr. Robert E. Brown
Senior Vice President, Regulatory Affairs
GE-Hitachi Nuclear Energy Americas, LLC
3901 Castle Hayne Rd MC A-45
Wilmington, NC 28401

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 121 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION

Dear Mr. Brown:

By letter dated August 24, 2005, GE-Hitachi Nuclear Energy Americas, LLC (GEH) submitted an application for final design approval and standard design certification of the economic simplified boiling water reactor (ESBWR) standard plant design pursuant to 10 CFR Part 52. The Nuclear Regulatory Commission (NRC) staff is performing a detailed review of this application to enable the staff to reach a conclusion on the safety of the proposed design.

The NRC staff has identified that additional information is needed to continue portions of the review. The staff's request for additional information (RAI) is contained in the enclosure to this letter.

To support the review schedule, you are requested to provide the requested additional information within 45 days of the date of this letter.

If you have any questions or comments concerning this matter, you may contact me at 301-415-5787 or rdf@nrc.gov, or you may contact Amy Cubbage at 301-415-2875 or aec@nrc.gov.

Sincerely,

/RA/

Rocky D. Foster, Project Manager
ESBWR/ABWR Projects Branch 2
Division of New Reactor Licensing
Office of New Reactors

Docket No. 52-010

Enclosure: Request for Additional Information

cc: See next page

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Rocky D. Foster, Project Manager
ESBWR/ABWR Projects Branch 2
Division of New Reactor Licensing
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Docket No. 52-010
Enclosure:
1. Request for Additional Information
cc: See next page
Distribution: See next page

ACCESSION NO. ML073230814

OFFICE	NGE2	NGE1	SLBP/BC	NGE1/BC
NAME	RFoster	SGreen	HHamzehee	MShuaibi
DATE	11/19/2007	11/20/2007	11/27/2007	12/5/2007

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SUBJECT: REQUEST FOR ADDITIONAL INFORMATION LETTER NO. 121 RELATED TO ESBWR DESIGN CERTIFICATION APPLICATION dated December 5, 2007

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Requests for Additional Information (RAIs)
ESBWR Design Control Document (DCD), Revision 4

RAI Number	Reviewer	Question Summary	Full Text
19.1-13, Supplement No. 1	Fuller E	Update the response to NRC RAI 19.1-13 to be current with Revision 2 of the PRA.	Please update the response to NRC RAI 19.1-13 to be current with Revision 2 of the Probabilistic Risk Assessment (PRA). In addition, please provide a discussion on possible changes to the various release category source term magnitudes that could result from external events and shutdown severe accidents, as compared to the values calculated for the full power internal event accidents.
19.1-154	Pohida M	Justify why the Circulating Water and the Feedwater and Condensate systems were screened from the Shutdown Flooding PRA	Please justify in the PRA why the Circulating Water System and the Condensate and Feedwater systems were screened as potential flooding sources during shutdown. In operating BWRs, Condensate and Feedwater systems are run in the cleanup mode after maintenance. In operating BWRs, circulating water is often run three (3) days before reactor startup.
19.1-155	Caruso M	Additional information is needed to assess the technical adequacy of the ESBWR PRA	Section 19.2.1 of the ESBWR Design Control Document (DCD), Revision 4, states: "Where applicable, ASME-RA-Sb-2005 (References 19.2-2 thru 19.2-4) capability Category 2 attributes are included in the analysis". In order to judge the technical adequacy of the ESBWR PRA documented in NEDO-33201, Revision 2, additional information is needed that describes the extent to which the ESBWR PRA incorporates capability Category 2 attributes of the ASME Standard for PRA (i.e., ASME-RA-Sb-2005, hereafter referred to as "the Standard") as described below: A. Please identify those high level requirements or capability Category 2 attributes of the Standard that have not been embodied in the ESBWR PRA. If a requirement or attribute has been omitted because it is considered inapplicable to a PRA developed in support of a Design Certification, please justify the omission.

RAI Number	Reviewer	Question Summary	Full Text
			B. For those high level requirements or capability Category 2 attributes of the Standard that are considered applicable, but have not been incorporated, please address the impact of not including them on the qualitative and quantitative results of the PRA. C. If a self-assessment or peer review process has been performed for the ESBWR PRA, please describe this process and the facts and observations that came out of it.
19.2.4-1, Supplement No. 1	Fuller E	Provide severe accident management program discussion or commitment.	The response to RAI 19.4.2-1 in MFN 05-169 indicated that Revision 1 to the DCD Chapter 19 would contain a list of COL Applicant commitments that would include text indicating that the COL applicant referencing the certified design will develop and implement severe accident management guidance, along with the required procedures and training, using the framework provided in DCD Chapter 18, Appendix A. In Revision 4 of the DCD, such text does not exist. Instead, Chapter 18, Revision 4, now has the following wording: Technical bases for severe accident management (core damage prevention and mitigation strategies and actions to limit radionuclide releases with off-site dose limits) are documented in Item 7 of DCD Tier 1, Table 3.3-1 for HFE. Standard guidelines, procedures, and training modules are developed as described in Reference 18.1-1. The PRA and Human Reliability Assessment (HRA) confirm that the Emergency Procedure Guidelines (EPGs) and severe accident guidance effectively address: • Preventing core damage, • Recovering from core damage • Maintaining containment integrity, and • Minimizing radionuclide releases The standard guidance and EPGs are used to develop and validate site-specific severe accident mitigation guidelines and procedures that satisfy Reference 18.1-2.

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			Reference 18.1-1, dated July 2007, is an ESBWR Licensing Topical Report that describes the Man-Machine Interface and Human Factors Engineering Implementation Plan. Reference 18.1-2 is an Industry document (NEI 91-04, Revision 1) on Severe Accident Closure Guidelines that provides an overview on how the existing plants should implement severe accident management guidelines. Section 3.2.4.5 of Reference 18.1-1 describes what GE calls its Emergency Management Program. This section lists GEH and applicant responsibilities in developing an emergency management program that would include procedures for preventing and mitigating the effects of severe accidents. The tone of the write up suggests that GEH may be considering the accident management program to be a COL information item. However, Section 19.3.6 of Revision 4 of the DCD omits any mention of it. In light of the above, please answer the original RAI again, namely, provide a discussion or commitment (combined operating license information item) regarding the severe accident management program under which guidance and training would be provided on the use of such features as containment venting, drywell sprays, and AC-independent fire pumps for isolation condenser make-up.
19.2-24, Supplement No. 2	Fuller E	Provide an analysis that demonstrates no core debris would reach the sumps.	The response to Supplement 1 of RAI 19.2-24 states that a design assumption is that narrow drain channels into the sump would be sufficiently long to be plugged by freezing melt prior to reaching the sump. Upon completion of the detailed design, please provide an analysis that demonstrates that the melt would indeed freeze and plug the channels, and that no molten core debris would reach the sumps. In addition, please explain how, if molten core debris would reach the sump, radioactivity would be released to the environment.

RAI Number	Reviewer	Question Summary	Full Text
19.2-23 S02 and 19.2-25 S02	Fuller E	Provide BiMAC test results.	In MFN 07-013 Supplement 3 and MFN 06-313 Supplement 5 it was stated that BiMAC test results would be provided in September 2007 in NEDO-33201 Section 21. The results were not, in fact, provided in that submittal. Until the results are reviewed and the staff concludes that they provide assurance that the BiMAC design is appropriate, the subject RAIs cannot be resolved. Please provide complete documentation of the BiMAC test results.
19.2-41, Supplement 2	Xu J	Provide adequate justification for imposing temperature condition for the drywell head at 43.3^o C (110° F) while maintaining the drywell airspace at 260° C (500° F) steady state temperature	The staff reviewed GEH's response (MFN 06-268 S1, dated April 13, 2007), and seek clarification on Response c) as follows: (A) The staff determined that the justification for the low temperature boundary condition for the drywell head (43.3^o C or 110°F) under the accident condition is insufficient. In the original containment fragility analysis as documented in GEH PRA report, Revision 1, Appendix B.8, GE specified the accident steady-state temperature of 260° C (500° F) for the drywell airspace and drywell head shell (neglecting the cooling effect of the pool above the head). During the staff's February 5-7, 2007 on-site audit, GEH informed the staff that GEH will replace the existing containment performance analysis with a new and more technically enhanced 3-D ABAQUS/ANACAP-U finite element analysis. Based on GEH's presentation of the new fragility analysis ABAQUS/ANACAP-U model, the staff considered the approach acceptable. However, the staff identified an issue with the new ABAQUS/ANACAP-U analysis concerning the temperature boundary condition of 43.3^o C (110° F) imposed for the drywell head shell, as opposed to the assumption of 260° C (500° F) for the drywell head shell made in the original containment performance analysis. GEH indicated at the staff's February 5-7, 2007 on-site audit meeting that the pool water directly above the drywell head shell will keep the head shell at 43.3^o C (110° F). Since the drywell head airspace is only separated from the drywell air space by the bellow which is made of a steel plate, the staff questioned whether the head shell can be kept at 43.3^o C (110° F) while the drywell air space is assumed to be at 260° C (500° F).

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			Consequently, the staff requests the applicant to submit a technically sound and complete justification, possibly through an appropriate heat transfer analysis, for its assumption that the drywell head remains at the normal operating containment atmosphere temperature of 43.3^{o} C (110°F) under the accident condition, while the containment atmosphere is at 260° C (500°F) at steady state. (B) In addition, the staff determined that the statement in the response, "A hotter inner surface temperature helps resist the separation of the flanges on the inner surface because the hot inner surface must expand relative to the colder outer surface. Because the A516 material does not have any significant degradation in ultimate strength or ductility up to this temperature of 260°C (500°F), there is no change in the failure mode. This means it requires a somewhat higher pressure to overcome this thermal induced stress to reach the same flange distortion level to yield the bolts," requires clarification, and that additional information on the response of the bolted flange connection at elevated temperature is needed. The staff agrees that the above statement may be applicable early in the temperature transient, when there is a significant temperature gradient from the inside surface to the outside surface of the flanges. However, as the transient progresses and the temperature of the bolted flange connection reaches a steady state condition, the effect of elevated temperature may reduce the pressure at which leakage is predicted. The applicant has not addressed this in its response. Consequently, the staff requests the applicant to: (1) describe in detail the steady state temperature condition of the bolted flange connection for the accident condition;

RAI Number	Reviewer	Question Summary	Full Text
			(2) identify the coefficients of thermal expansion for the flange material and the bolt material; (3) discuss o-ring degradation at the accident temperature; (4) provide a plot showing the average temperature of the flanges and the average temperature of the bolts, from initiation of the transient up to achievement of the steady state condition; (5) describe the stress and deformation state of the bolted flange connection at the steady state condition; and, (6) provide the estimate of pressure capacity of the bolted flange connection at the steady state condition and technical basis for the estimate. (C) GEH indicated at the staff's February 5-7, 2007, on-site audit that the bolt preloads for the major penetrations will be set for the ultimate containment pressure capacity. However, the new containment performance analysis using ABAQUS/ANACAP-U as documented in the proposed DCD Tier 2, Appendix 19B, did not calculate the containment Level-C pressure capacity; rather it demonstrated that the containment Level-C pressure capacity will be higher than the pressure loads induced by the more likely containment accident scenarios and the 100% MWR. Therefore, GEH needs to stipulate quantitatively in Tier 2* the bolt preloads for the major penetrations including the drywell head, which were used in the ABAQUS/ANACAP-U analysis.

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19.2-80	Fuller E	Provide additional discussions pertaining to Section 11.3.4.2 of Revision 2 of the PRA	The focused Level 2 results in Table 11.3-26 of Revision 2 of the PRA show a higher value for OPW2 than for BYP. A discussion of the reasons for the higher value for OPW2 as compared to BYP has not been provided. Please provide additional discussions to augment those in Section 11.3.4.1 ("Focus Level 2 Internal Events") for OPW2 along with BYP as major effects. In addition, the BiMAC device is not safety-related but it is important for some potential containment failure modes. Please provide a discussion on the omission of specific reference to BiMAC in the RTNSS and the Focus Level 2 sensitivity studies.
19.2-81	Fuller E	Provide additional DCH analysis.	The ROAAM analyses for DCH show short periods of potentially very high temperatures in the LDW atmosphere (up to 4000 K). Chapter 21 of the PRA states that "*These, and the presence of potentially large quantities of melt in the LDW, indicate that the LDW liner could be subject to local failures, a condition that is noted in our HP CPET and is accounted for in Level-3 PRA.*" GEH has also stated that liner failure in the LDW space would not constitute containment failure because of the presence of structural "lips" that provide isolation of the gap space from that of the upper portions of the containment wall. Basically, it says that this release path would be negligible. No comment is made in PRA Revision 2 Section 8 about DCH LDW liner failure. Section 9.5 notes that local damage to the liner in the lower drywell will be studied as a sensitivity case in Section 11 and as such no DCH sequence is selected for the baseline case. Appendix 9A contains a "DCH" case which appears to be a basemat attack accident with overpressure failure due to non-condensables – not local LDW liner over-temperature failure. Several tables in Sections 10 and 11 note a DCH sensitivity set of results. This DCH has a frequency of 2.56×10^{-12} per reactor-year but no source for this frequency is quoted, and neither is the source of the release fractions given. Please provide the analysis that demonstrates that the DCH-induced LDW local liner failure is a negligible release path.

RAI Number	Reviewer	Question Summary	Full Text
19.2-82	Fuller E	Provide more detailed information pertinent to offsite consequence evaluations.	The offsite consequence evaluation (Level 3 analysis) in Chapter 10 of NEDO 33201- ESBWR certification PRA indicates the use of both generic and ESBWR specific data. The analysis uses MACCS2 Version 1.13.1 computer code over a range of possible weather conditions and accident specific assumptions. The generic input parameters are from "Sample Problem A" of the MACCS2 volume 1, with the assumption of uniform population density based on the Sandia Siting Study (Ref. 1), a meteorology condition comparable with ALWR URD (Ref. 2), and an assumption of ground release with no evacuation, relocation, or sheltering during reactor release. The ESBWR specific data are plant performance analyses from containment event tress (Chapter 8), source terms for select sequences (Chapter 9), and the projected core inventory at the time of accident. The analysis also includes a sensitivity analysis on the meteorology, and assumptions regarding release elevation and buoyancy. The analysis concludes that the selected parameters and assumptions would result in bounding consequences. Review of the methods and assumptions has identified a number of questions related to the details of the GEH analyses and conclusions that follow. 1. As indicated in the analysis, the meteorology and the population density are site specific. Please elaborate the assumptions that the use of ALWR URD meteorology, and the Sandia Siting Study population density (1980's vintage data) would lead to bounding results. Note that the annual disturbances in meteorology could lead to variations in offsite consequences of the order of about ±20 percent. 2. Tables 10.4-1a, and10.4 -1b provide individual risk (0-1 mile) and is defined as the total number of early fatalities within one mile divided by the total population within one mile. The assumption of fully populated first segment (0-1 mile) would lead to an underestimation in the stated risk. Please provide additional information on early fatality and latent cancer estimates for the 0-1 and 0-10 miles. Note that, given the assumption of

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			ground release, and no evacuation, relocation, or sheltering, the majority of doses would be to those residing within 10 miles of the plant. The estimated population doses in Tables 10.5-2a and 10.5-2b do not seem to correlate with the values given in Tables 10.4-1a and 10.4-1b. 3. Chapter 8 of NEDO 33201 provides information on containment performance. Table 8.2-2 provides CET release category frequencies which are similar, except for the OPVB and OPW1 release categories, to those used in Chapter 9, Table 9-1, and Chapter 10, Table 103-3b. The frequency values listed for the OPVB and OPW1 release categories in Table 8.2-2 are 1.6 E-11, and 3.2 E-11 per reactor-year, respectively. On the other hand, in the Chapter 9 tables, the frequency values listed for the same release categories are 6 E-12, and less than 1 E-15 per reactor-year, respectively. Please explain.
19.2-83	Fuller E	Clarify results of the vacuum breaker sensitivity study	The results discussion for the Vacuum Breaker sensitivity study in PRA R2 Section 11.3.2.3 "Vacuum Breakers" mentions a "*… significant increase in nTSL frequency of almost 2 orders of magnitude over the base Level 2 mode*l" whereas Table 11.3-18 shows an increase of just over a factor of two. Also, note that the referenced Tables 11.4-18 and 11.4-19 do not exist. The text in Section 11.3.2 appears to be incorrect. It is believed that the increase in frequency should be about a factor of 2 and not by 2 orders of magnitude (see Table 11.3-18 and 11.3-19). It appears that the references to Tables 11.4-18 and 11.4-19 are in error, and instead their references should be made to Tables 11.3-18 and 19 (i.e., the numbers in these tables match the text). Please clarify.
19.2-84	Fuller E	Clarify results of the Level 3 PRA sensitivity study	PRA R2 Section 10.5 (SENSITIVITY STUDY) states, "*Elevated release with and without buoyant plume energy rise is studied along with sensitivity on population density*". The results for varying population density are not apparent. Please clarify.

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19.2-85	Xu J	Provide justification for HCLPF calculations for the performance spectrum for both rock and soil sites	In NEDO-33201 Revision 2, Section 15, "Seismic Margin Analysis"(SMA), GEH provided a seismic fragility analysis which used a single performance-based design spectrum (PBDS), as shown in Figure 15-2. PBDS falls below the certified seismic design spectrum (CSDS) for frequencies below 9 Hz. Since SECY 93-087 clearly asks for the HCLPF capacity to be calculated with respect to the design basis SSE, which in the certified design means CSDS. The staff requests that GEH revises SMA using CSDS to calculate HCLPF capacities. In addition, during the staff on-site audit, February 5-7, 2007, GEH presented HCLPF calculations for rock and soil sites using two separate ground inputs. To facilitate the staff review of the fragility method that was applied to the single envelope spectrum, as presented in Section 15, please provide the details of the HCLPF calculations for the containment and for the control building, for the combined rock and soil spectrum. The staff's concern is related to the unconventional spectrum shape for the unified ground spectrum and needs to confirm that the HCLPF calculations for the unified spectrum are of the same good quality as the HCLPF calculations
22.5-17, Supplement No.1 (MFN 07-455, August 23, 2007)	Scarbrough T	Explain approach to evaluate potential RTNSS system interactions	In its response to RAI 22.5-17 in MFN 07-455 (August 23, 2007), GEH states that it evaluated potential adverse system interactions for nonsafety-related components from functional or spatial interactions. GEH reports that the evaluation did not identify any structures, systems, and components that should be considered for Regulatory Treatment of Non-Safety Systems (RTNSS). Explain how potential adverse system interactions for nonsafety-related components from functional or spatial interactions will be identified and addressed during detailed engineering and construction phase to ensure that the functions of safety related and RTNSS systems will not be adversely impacted. Please add a COL item if necessary to address this issue.

DC GE - ESBWR Mailing List

cc:

Ms. Michele Boyd
Legislative Director
Energy Program
Public Citizens Critical Mass Energy
and Environmental Program
215 Pennsylvania Avenue, SE
Washington, DC 20003

W. Craig Conklin, Director
Chemical and Nuclear Preparedness &
Protection Division (CNPPD)
Office of Infrastructure Protection
Department of Homeland Security
Washington, DC 20528

Mr. Marvin Fertel
Senior Vice President
and Chief Nuclear Officer
Nuclear Energy Institute
1776 I Street, NW
Suite 400
Washington, DC 20006-3708

Mr. Ray Ganthner
AREVA, Framatome ANP, Inc.
3315 Old Forest Road
P.O. Box 10935
Lynchburg, VA 24506-0935

Vanessa E. Quinn, Acting Director
Technological Hazards Division
National Preparedness Directorate
Federal Emergency Management Agency
500 C Street, NW
Washington, DC 20472

Email

APH@NEI.org (Adrian Heymer)
awc@nei.org (Anne W. Cottingham)
bennettS2@bv.com (Steve A. Bennett)
bevans@enercon.com (Bob Evans)
bob.brown@ge.com (Robert E. Brown)
BrinkmCB@westinghouse.com (Charles Brinkman)
chris.maslak@ge.com (Chris Maslak)
CumminWE@Westinghouse.com (Edward W. Cummins)
cwaltman@roe.com (C. Waltman)
dan1.williamson@ge.com (Dan Williamson)
david.hinds@ge.com (David Hinds)
david.lewis@pillsburylaw.com (David Lewis)
David.piepmeyer@ge.com (David Piepmeyer)
dlochbaum@UCSUSA.org (David Lochbaum)
don.lewis@ge.com (Don Lewis)
erg-xl@cox.net (Eddie R. Grant)
frankq@hursttech.com (Frank Quinn)
Frostie.white@ge.com (Frostie White)
gcesare@enercon.com (Guy Cesare)
george.honma@ge.com (George Honma)
george.stramback@gene.ge.com (George Stramback)
george.wadkins@ge.com (George Wadkins)
GovePA@BV.com (Patrick Gove)
greshaja@westinghouse.com (James Gresham)
gzinke@entergy.com (George Alan Zinke)
hickste@earthlink.net (Thomas Hicks)
james.beard@gene.ge.com (James Beard)
jcurtiss@winston.com (Jim Curtiss)
jgutierrez@morganlewis.com (Jay M. Gutierrez)
jim.kinsey@ge.com (James Kinsey)
jim.riccio@wdc.greenpeace.org (James Riccio)
JJNesrsta@cpsenergy.com (James J. Nesrsta)
joel.Friday@ge.com (Joel Friday)
john.o'neil@pillsburylaw.com (John O'Neil)
john.sorensen@ge.com (John Sorensen)
Joseph.savage@ge.com (Joseph Savage)
Joseph_Hegner@dom.com (Joseph Hegner)
junichi_uchiyama@mnes-us.com (Junichi Uchiyama)
kathy.sedney@ge.com (Kathy Sedney)
kenneth.ainger@exeloncorp.com (Kenneth Ainger)
KSutton@morganlewis.com (Kathryn M. Sutton)
kurt.schaefer@ge.com (Kurt Schaefer)
kwaugh@impact-net.org (Kenneth O. Waugh)
lou.lanese@ge.com (Lou Lanese)
lynchs@gao.gov (Sarah Lynch - Meeting Notices Only)
MaddenG@BV.com (George Madden)
Margaret.Bennet@dom.com (Margaret Bennet)

maria.webb@pillsburylaw.com (Maria Webb)
mark.beaumont@wsms.com (Mark Beaumont)
Marvin.Smith@dom.com (Marvin L. Smith)
matias.travieso-diaz@pillsburylaw.com (Matias Travieso-Diaz)
media@nei.org (Scott Peterson)
mgiles@entergy.com (M. Giles)
mike_moran@fpl.com (Mike Moran)
mwetterhahn@winston.com (M. Wetterhahn)
nirsnet@nirs.org (Michael Mariotte)
PAC2@nrc.gov (Peter Cochran)
pareez.golub@ge.com (Pareez Golub)
Pat.Woodfin@ge.com (Pat Woodfin)
patriciaL.campbell@ge.com (Patricia L. Campbell)
paul.gaukler@pillsburylaw.com (Paul Gaukler)
Paul@beyondnuclear.org (Paul Gunter)
peter.jordan@ge.com (Peter Jordan)
phinnen@entergy.com (Paul Hinnenkamp)
pshastings@duke-energy.com (Peter Hastings)
RJB@NEI.org (Russell Bell)
RKTemple@cpsenergy.com (R.K. Temple)
roberta.swain@ge.com (Roberta Swain)
ronald.hagen@eia.doe.gov (Ronald Hagen)
sandra.sloan@areva.com (Sandra Sloan)
SauerB@BV.com (Robert C. Sauer)
sfrantz@morganlewis.com (Stephen P. Frantz)
sharon.lyons@ge.com
steven.hucik@ge.com (Steven Hucik)
steven.stark@ge.com (Steven Stark)
tom.miller@hq.doe.gov (Tom Miller)
trsmith@winston.com (Tyson Smith)
VictorB@bv.com (Bill Victor)
waraksre@westinghouse.com (Rosemarie E. Waraks)
wayne.marquino@ge.com (Wayne Marquino)
whorin@winston.com (W. Horin)