The RETRAN & VIPRE Newsletter

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INSIDE

April 2006 RETRAN and VIPRE User Group Meeting Held in Salt Lake City

The spring 2006 RETRAN/VIPRE User Group (RUG) Meeting was held at the Red Lion Hotel in Salt Lake City, Utah. Mr. Gregg Swindlehurst, the RVUG Steering Committee Chairman, welcomed the attendees and opened the meeting, which was attended by EPRI, nine U.S. utilities, one U.S. commercial vendor, and CSA. AEP, AmerenUE, Dominion, Duke Power, Entergy, Florida Power & Light, Southern Nuclear, STP Nuclear Operating

Jong Chang of Entergy Attended the April UGM in Salt Lake

Co, TXU and Westinghouse were each represented.

The financial status of RETRAN project was presented by CSA, including a summary of the revenues and expenses for 2005 and projections for 2006.

A summary of the recently completed and future work scope tasks was presented by CSA. A new release of RETRAN-3D, MOD004.2, had been transmitted to RUG member organization in mid-May. Some discussion was given regarding new features included in the version and significant error corrections that may have some impact on results. New user guidelines were also included in the User Manual (Volume 3) for use of the slip model, the five-equation model, the enthalpy transport model, and the bubble rise model.

The status of unresolved trouble reports for both RETRAN-02 and RETRAN-3D were discussed with a plan for resolving them.

CSA also presented a similar summary of the VIPRE project. CSA identified potential VIPRE development tasks for 2006 and members were asked to provide additional work scope items of interest so the 2006 work scope can be finalized.

A focus group was formed from the VUG membership to identify specific visualization needs that should be addressed. Anyone interested in participating should contact Gregg Swindlehurst or Mark Paulsen.

Member organizations summarized RETRAN and VIPRE activities.

- Westinghouse gave a summary of ongoing activities for the VIPRE-W code. They included approval of Westinghouse methodologies using VIPRE-W, recent applications, and ongoing development for Westinghouse fuel.

- CSA summarized work Performed for Duke Energy to analyze the MIST Facility SBLOCA Test Analysis. The purpose of the work was to demonstrate RETRAN-3D's ability to simulate boiler-condenser mode heat transfer needed for core cooling during certain accident scenarios.

- Entergy presented a summary of RETRAN-3D related analysis work that has been performed recently. They included Appendix R analysis, steam generator tube rupture analysis, power uprate support, IP2 & IP3 plant trip benchmark, and IP2 & IP3 Simulator Benchmark.

April 2006 RETRAN and VIPRE User Group Meeting Held in Salt Lake City (Cont'd)

- Duke presented NSRR RIA Test Thermal Response Modeling. Duke has been working with EPRI and CSA to try to develop a method to quantify the thermal response caused by fuel expulsion by benchmarking a model of several test capsules against test data.

- CSA gave a summary of RETRAN-3D Reactivity Insertion Sensitivity Studies in support of the EPRI Fuel Reliability Program. The analyses are intended to identify modeling sensitivities with respect to high burn-up fuel assemblies during RIA transients.

- Westinghouse presented an overview of the RETRAN-02 Superheat Model Development Status and the challenges associated with modeling superheat in steam generators that are drying out.

- Westinghouse also gave a discussion on a new asymmetric SG plugging model that allows the user to define initial asymmetric flow conditions was given.

Daren Chang, STPNOC, was elected to the Steering Committee as a VIPRE representative. Dave Huegel of Westinghouse will continue to serve on the Steering Committee for RETRAN issues.

Current RVUG Steering Committee members are:

Serhat Lider, Westinghouse discusses RETRAN Superheat Models

Gregg Swindlehurst, Duke (Chairman)
Andres Gomez, Iberdrola
Sama Bilbao y Leon, Dominion
David Huegel, Westinghouse (RETRAN)
Adi Irani, Entergy Nuclear Northeast
Daren Chang, STPNOC (VIPRE)

The fall RVUG meeting will be held at the Dominion facilities in Richmond, Virginia, on November 6-7.

RIA Investigations Reported at the 2006 UGM

Gregg Swindlehurst, Duke Energy

At the April 2006 meeting, two presentations were made relating to investigation of fuel rod failure limits and high burnup fuel. Experiments on fresh or low burn-up fuel conducted in the 1970's supported the current peak fuel enthalpy limit of between 200 and 280 cal/gm. However, tests using high burnup fuel performed in Japan and France in the early 1990's suggested that these limits may be too high.

In the newer tests, fuel rod failure occurred at fuel enthalpy limits that were well below that of unirradiated fuel. It was also observed that hot fuel particles were dispersed into the coolant associated with a corresponding pressure pulse. Based on these experimental results, the NRC has proposed lower fuel enthalpy limit values. The industry has resisted this lower limit and has been involved in activities that demonstrate that the consequences of fuel rod failure are not as severe as that implied by the experiments.

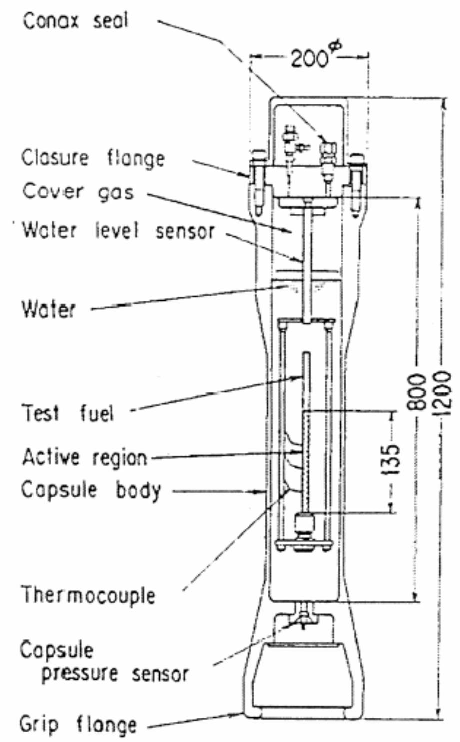
The presentation made by Duke Energy describes RETRAN-3D modeling methods for two of the Japanese experiments with the goal of determining how the thermal response caused by the fuel expulsion can be quantified.

The CSA presentation focused on RETRAN calculations to help evaluate the mechanical energy and pressure response from the fuel-coolant interaction following the fuel dispersal.

NSRR RIA Test Thermal Response Modeling

By Gregg Swindlehurst and Dewain McClain, Duke Energy

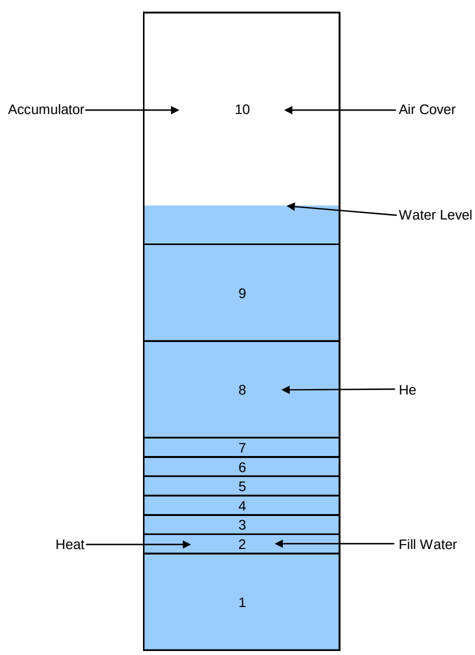
The PWR rod ejection accident initiating from zero power is the most severe RIA licensing basis event when considering the cal/gm regulatory acceptance limit. For large ejected rod worths the rapid expansion of the fuel pellet can overstress the cladding and cladding failure can occur. Failure of the cladding will release the gas inventory in the fuel rod. For very high ejected rod worths some of the fuel can be dispersed through the crack in the cladding in the form of pellet fragments and powder, and additional fission gas can be released from the fuel pellet grain boundaries. The release of gas from the fuel rod, and the rapid generation of steam from the heat transfer associated with dispersed fuel contacting the coolant, will cause an increase in the pressure of the coolant. In addition the mechanical energy associated with pressurizing the coolant can raise concerns of the potential to damage fuel assemblies and the primary pressure boundary. A large unknown in this scenario is the rate of heat transfer from the dispersed fuel to the coolant, and to what extent the heat transfer causes boiling vs. sensible heating of the coolant.



Duke Energy has been working with EPRI and CSA to develop models and methods to quantify the thermal response caused by RIA fuel expulsion. Duke is focusing on what can be learned from Nuclear Safety Research Reactor (NSRR) tests TK-2 and JMH-5. Anatech has provided information on the NSRR test facility and specifications for the tests of interest, but some test parameters were unavailable. Duke has developed RETRAN-3D simulation models of the facility in an attempt to benchmark the models and to develop insight into what can be learned and applied to the regulatory issue.

NSRR Tests

RIA tests conducted in the NSRR in Japan include data that can be used to gain insights into heat transfer resulting from dispersed fuel and gas release following cladding failure. The TK-2 and the JMH-5 tests are used in this evaluation. Both tests involve a short section of a fuel rod in a test capsule submerged in 70 °F water at atmospheric pressure with an air cover gas (adjoining figure). The fuel rod is subjected to a significant power pulse that results in cladding failure and partial fuel expulsion. The experiment instrumentation captures the thermal response of the test capsule, including the movement of the water level and the pressure response. Post-test measurements include the mass of dispersed fuel and the mass of fission gas released. The duration of the power pulses are approximately 10 msec.



RETRAN MODEL

The RETRAN-3D model of the NSRR facility was constructed to capture the features of both experiments (See Figure). Model highlights include:

- RETRAN-3D MOD 4.2
- 5-equation option and non-condensable gas flow option
- Maximum timestep = 0.05 milli-second
- Volume 10 uses the accumulator model with 10 cm of air over 2 cm of water
- Volumes 8 & 9 are each 5 cm high
- Volumes 3-7 are each 1 cm high
- Volume 1 (30 cm high) is closed off with a valve at 1 msec to prevent flow of 70 °F water into heated volume
- Helium injected into Volume 8 to model release of fission gases
- Heat is added to Volume 2, which is sized to allow steam generation, to model the heat added by the dispersed fuel fragments

Capability of RETRAN-3D to Simulate NSRR RIA Tests

The NSRR RIA tests present several modeling challenges for the RETRAN-3D code. The tests were conducted with initial conditions of stagnant 70 °F water at atmospheric pressure. Following the power burst, a rapid heating of the water causes boiling in one millisecond and the volumetric expansion due to phase change caused by boiling at low pressure is very dramatic. The boiling process occurs simultaneously with the condensation process. The extent of mixing with the water in the test capsule as the fuel is dispersed into the coolant is unknown. Non-condensable gas is also released upon cladding failure. The ultimate objective of quantifying the rate of heat transfer during the ten millisecond duration of the test is complicated by all of these considerations.

NSRR RIA Test Thermal Response Modeling (Cont'd)

The difficulty of representing the test facility and selecting the code options became readily apparent once the hand calculations originating from the test data were performed. Only a limited percentage of the water in the test capsule was boiling - if the deposited energy from the dispersed fuel fragments was evenly mixed with all of the test capsule water no boiling would result. So, all of the deposited energy had to be modeled as heating a small volume of water for boiling to occur. Another modeling concern was that the expanding steam would contact subcooled water and condense unless special modeling was used. The five-equation model option was necessary for the steam to be able to flow into adjacent subcooled volumes without instantaneously condensing. A final problem was that the occurrence of boiling would push heated liquid out of the volume being heated, causing the volume to dry out. The deposited energy in the displaced heated liquid would not cause additional boiling. Injection of saturated liquid via a fill junction into the heated volume was attempted to compensate for this modeling limitation. RETRAN-3D was able to model the effect of non-condensable gas injection due to cladding failure, and the transport of that gas, although not the actual fission gas components.

The overall result of the RETRAN-3D modeling effort was that, although the general progression of the NSRR tests was predicted, the limited mixing associated with the local boiling of the water in the test capsule could not be accurately simulated with the one-dimensional modeling approach used. The RETRAN-3D predictions resulted in excessive mixing of the locally heated water with the bulk subcooled water inventory in the test capsule. This was mainly a result of the expansion of the steam pushing heated water into the adjacent volume and that heated water not being further heated to boiling. A more sophisticated modeling approach would be required to simulate these effects.

Applicability of NSRR Test Results to PWR HZP Rod Ejection Event

The NSRR Tests are conducted with a single fuel rod segment in 70 °F water and stagnant (no flow) conditions. The fuel rod to water volume ratio on a percentage basis is 0.8% for Test TK-2 and 2.2% for test JMH-5. The subcooling is 142 °F, and the enthalpy change to achieve boiling is 1112 Btu/lbm. For a PWR core at hot zero power the fuel rod to water volume is 47% (OD = 0.374 in and pin pitch = 0.496 in), the initial subcooling is 93 °F, and the enthalpy change to achieve boiling is 1083 Btu/lbm. The coolant velocity is 16 ft/sec, which means that the coolant can travel 2 inches in the 10 msec of interest. This 2 inches of additional coolant during the 10 msec duration of the event is not significant. Only the fuel rod to water volume parameter is significantly different. For Test TK-2, if the fuel rod to water volume was increased from 0.8% to 47%, the coolant temperature increase would be only 6 °F. For Test JMH-5, if the fuel rod to water volume was increased from 2.2% to 47%, the coolant temperature increase would be 165 °F, which would cause additional boiling. However, the energy deposition in the failed fuel for Test JMH-5 was 283 cal/gm. For a more reasonable but still conservative energy deposition value of 140 cal/gm, which is the Test TK-2 value, the temperature increase would be only on the order of 80 °F, and no additional boiling is predicted.

On this basis the NSRR test results would be expected to be representative for the PWR HZP rod ejection event for the amount of boiling that would be expected per unit length of failed cladding.

RETRAN-3D RIA Fuel Dispersal Model Sensitivity Studies

CSA has been involved in an EPRI sponsored RIA analysis effort for the EPRI Fuel Reliability program. The EPRI project is directed at identifying and resolution of issues with respect to high burn-up fuel assemblies during RIA transients.

The current study involved RETRAN-3D MOD004.1 model development to compute the effect of failed fuel energy dispersal during a RIA. The work focused on the sensitivity of the pressure response in the core and plenum regions to RETRAN-3D modeling methods.

Background

A previous scoping study (performed in 1996) using RETRAN-3D showed that significant pressure increases can be generated in the core regions by the dispersion of the failed fuel energy into the fuel channel. The energy was simulated as short duration (10ms to 100ms) pulses.

In the older analysis, the failed fuel dispersal energy was lumped into the average core region. There was a concern that averaging the failed fuel into the larger non-failed regions was not a correct approach since local phenomena would not be captured.

The current work, involved the development of RETRAN-3D models were to study the impact of modeling detail in the failed fuel regions on the RIA pressure response in the core and plenums.

Summary

Two RETRAN-3D core models were used to study the effects of fuel dispersal energy deposition during an RIA. The Base model accounted for failed fuel as a single, averaged channel and the remaining non-failed assemblies as the second channel. The Sensitivity model was used to study the pressure response to adding modeling detail in the core region representing the failed fuel assemblies. The multiple channel Sensitivity model accounted for five failed assemblies as separate channels. These were connected to 12 additional adjacent non-failed channels. The effects of cross flow between assemblies and to the remaining large average core were included.

The overall system pressure response for regions such as the lower and upper plenums and average non-failed core regions are very similar between the two models.

The multiple channel model allows one to account for small scale flow patterns in the core. Where the issue of local pressure response may be important, the multiple channel model should be used. The Sensitivity model resulted in local pressures that are significantly different from the Base Model.

RETRAN-3D RIA Fuel Dispersal Model Sensitivity Studies (Cont'd)

Significant channel voiding occurred in both models, which was not observed in the older lumped core model.

The response of the energy dispersal is characterized by:

- Nearly Instantaneous Pressure Response
- Large Pressure Spikes
- Rapid Void Formation/Flashing
- Local Flow Reversals in Axial and Cross Flow Junctions
- Local Critical Flow in Some Junctions

Future Studies
The energy source term for these studies was modeled using a RETRAN-3D non conducting heat exchanger. This approach resulted in nearly instantaneous energy transfer to the coolant, .
and there was no accounting for heat transfer from the fuel particle surface. This means that the dispersed fuel is immediately cooled by the local fluid as it is released. This assumption is overly conservative since the dispersed fuel particle cooldown may be slower due to the time constant for heat transfer between the particles and the bulk fluid. In addition, the model assumes that the fuel particles remain in the dispersal volume when in fact the particle may be carried to adjacent regions containing lower temperature coolant.

Future studies have been proposed to examine these assumptions by developing a model to consider the fuel particle cooldown rate assuming that film boiling occurs at the particle surface.

VIPRE-01 Used to Generate Input Files for BOA

Many of the new requests for the VIPRE-01 code are due to the need to generate input files for the Boron-induced Offset Anomaly (BOA) Risk Assessment Tool. The EPRI BOA code has been developed to assess fuel behavior for Axial Offset Anomaly (AOA) risk.

The occurrence of PWR AOA is a limiting operational condition preventing many PWRs from operating with efficient core designs. AOA occurs when boron incorporates in corrosion products deposited in the steaming regions of high-duty fuel assemblies causing the reactor neutron flux to become skewed. AOA has affected operating cycles of a number of U.S. and international PWRs.

The AOA phenomenon is believed to result from several interrelated aspects. Subcooled boiling on the upper spans of high-duty assemblies causes enhanced corrosion product deposition. As this corrosion deposits thicken, boron is incorporated in the deposits causing a depression in neutron flux.

The BOA code has thermal-hydraulics models to calculate steaming rates on clean wall surfaces and within porous corrosion deposits. A system mass balance approach is used to model corrosion product release and deposition. Boron deposition processes have been incorporated based on both precipitation and physic-sorption phenomena. The BOA code also incorporates a PC Windows-based Graphical User Interface to easily input data and review output results.

The BOA code requires input of the local heat fluxes and fluid conditions. These local conditions are obtained from the VIPRE-01 code. Using power profiles from physics analyses, VIPRE-01 analyses are performed for several points during the fuel cycle. The VIPRE-01 input model uses the same geometric nodalization as the BOA code. Most analyses are performed with a quarter-core representation (four channels per assembly) and a uniform axial nodalization of 1 to 2 inches. The fine axial mesh is needed to accurately represent the effect of mixing vane grids on the heat transfer in the BOA code. Stacked cases are used to model the changes in power distribution during the cycle. A VIPRE-01 input option (IKEN=2, OPER.1) is activated to generate BOA AOA files containing the local pressure, temperature, mass flux, and density for each node in the model. Ten to fifteen files are generated to represent the local conditions for the various point in time needed to describe the complete fuel cycle.

RETRAN Training Sessions Draw Students Worldwide

The June RETRAN session at CSA involved 16 individuals from nine organizations representing a good cross section of the RETRAN user community. These were:

Thomas E. Ander, American Electric Power
Jeffrey D. Shelton, AmerenUE
Joseph R. Otero, CSA
Matthew Cameron, Duke Energy
Adam W. Strange, Duke Energy
Su Hyun Hwang, FNC Technology Co.
Kyung Jin Lee, FNC Technology Co.
Jong-Beom Lee, Korea Hydro & Nuclear Co., Ltd.
Chan-Su Jang, Korea Nuclear Fuel Co.
Gyo-Seob Lee, Korea Nuclear Fuel Co.
Dang Ho, Southern Nuclear
Stanley G. Cheyne, Westinghouse
Andrew Detar, Westinghouse
Julie Gorgemans, Westinghouse Belgium
Alan J. Macdonald, Westinghouse
Derek Seaman, Westinghouse

Congratulations to all of the new RETRAN training graduates.

RETRAN-3D MOD004.2 Now Available

RETRAN-3D MOD004.2 is now available and has been transmitted to RUG members. It is supported on HP, IBM, and SUN UNIX workstations and Windows XP platforms. Revision 6.2 to Volumes 1, 2, and 3, and Revision 6.1 to Volume 4 were included with the transmittal.

Twenty-eight modifications were made to RETRAN-3D MOD004.1 to create the latest version, RETRAN-3D, MOD004.2. The modifications included 24 error corrections and four modifications that added new optional features. A description of the modifications is available on the CSA website, www.csai.com.

RETRAN-3D User Group members can request RETRAN-3D MOD004.2 by contacting Pam Richardson, CSA, via email at pam@csai.com or by calling (208) 529-1700, Ext. 11.

Utility Enhances RETRAN-02 Trip Summary

RETRAN-3D has a feature that allows users to add description labels to trip cards, which then appear in the RETRAN-3D trip summary file. It's a convenient feature that one forgets about until..... you run a RETRAN-02 case and find that it's not available in the code.

Admittedly, it's a small feature, but we missed it when we used RETRAN-02. So CSA developed a simple utility using AWK that helps. This utility will scan the RETRAN-02 output file...grab comment cards from the trip data input segment and append them to the associated trips in the TRIP summary.

Here is an example from the RETRAN-02 UCRW sample problem.

Suppose the trip cards (a subset) look like this:

```
*            IDTRP IDSIG  IX1  IX2     SETPT     DELAY
040070   16        6       18    0      42.45       1.0     * HIGH PRESSURIZER LEVEL
040080   16       12        6    0        0.0       1.0     * LOW PRESSURIZER PRESS
040090   16        4       18    0     2425.0       1.0     * HIGH PRESSURIZER PRESS
040100   16        2        0    0       1.18       0.5     * NUCLEAR OVERPOWER
040110   16       14       -9    0        0.0       2.3     * OVER TEMPERATURE DELTA T
040120   16       14      -16    0        0.0       2.3     * OVER POWER DELTA T
040130   16       -9        9    0    24528.8       0.6     * LOW COOLANT FLOW ALL PUMPS
040140   16       12        4    0        0.0       1.0     * SCRAM FROM TURBINE TRIP
040150   16       12        5    0        0.0       0.5     * SCRAM FROM SIS
040340   20        1        0    0        0.0       0.0     * GENL MISC TRIP AT TIME  0.
040350   22       -4       18    0     2235.0       0.0     * PRESSUR HEATERS  CONTROL  ON
040360  -22       +4       18    0     2235.0       0.0     * PRESSUR HEATERS  CONTROL OFF
040410   16        1        0    0        1+9       0.0     * SCRAM ON TIME LOSS PUMP POWER
040420   25        1        0    0         0.       0.0     * URW REACTIVITY INSER
```

A typical RETRAN-02 trip summary will look like this.

```
TIME (SEC)  TRP ID  ACTION     REASON           SET POINT     DELAY (SEC)  IX1  IX2

0.000000E+00  20   GEN TRIP ON ELAPSED TIME OF  0.000000E+00  0.000000E+00   0  0
              25   GEN TRIP ON ELAPSED TIME OF  0.000000E+00  0.000000E+00   0  0
1.000000E-02  22   GEN TRIP ON LOW PRESSURE OF  2.235000E+03  0.000000E+00  18  0
2.350000E+00  16   GEN TRIP ON HIGH POWER   OF  1.180000E+00  5.000000E-01   0  0
2.650000E+00  22   RST TRIP ON HIGH PRESSUR OF  2.235000E+03  0.000000E+00  18  0
```

An application of the utility results in a trip summary that looks like this.

```
TIME (SEC)  TRP ID  ACTION     REASON           SET POINT     DELAY (SEC)  IX1  IX2

0.000000E+00  20   TRIP  ELAPSED TIME OF        0.000000E+00  0.000000E+00   0  0  GENL MISC TRIP AT TIME 0.
              25   TRIP  ELAPSED TIME OF        0.000000E+00  0.000000E+00   0  0  URW REACTIVITY INSER
1.000000E-02  22   TRIP  LOW PRESSURE OF        2.235000E+03  0.000000E+00  18  0  PRESSUR HEATERS  CONTROL ON
2.350000E+00  16   TRIP  HIGH POWER   OF        1.180000E+00  5.000000E-01   0  0  NUCLEAR OVERPOWER
2.650000E+00  22   RESET HIGH PRESSUR OF        2.235000E+03  0.000000E+00  18  0  PRESSUR HEATERS CONTROL OFF
```

The script has been placed on the RETRAN website for the convenience of RETRAN user group members http://www.csai.com/retran/summary.html. Download and execution instructions can be obtained there.

About This Newsletter

RETRAN Maintenance Program

The RETRAN/VIPRE Maintenance Program is a program that provides for the support of software developed and maintained by CSA. The main features of the Subscription Service include:

- the code maintenance activities for reporting and resolving possible code errors,
- providing information to users through the User Group Meetings and this newsletter, and
- preparing new versions of RETRAN and VIPRE.

The RETRAN Maintenance Program now has 19 organizations participating in the program, including 13 U.S. utilities and 6 organizations from outside of the U.S. Ten U.S. utilities are currently participating in the VIPRE maintenance program. A Steering Committee, composed of representatives from the participating organizations, advises CSA on various activities including possible enhancements for the code and the scheduling of future code releases. Information regarding the Maintenance Program can be obtained from:

Mark P. Paulsen
Computer Simulation & Analysis, Inc.
P. O. Box 51596
Idaho Falls, ID 83405
paulsen@csa.com or (208) 529-1700

Newsletter Contributions

The RETRAN/VIPRE Newsletter is published for members of the Subscription Service program. We want to use the newsletter as a means of communication, not only from CSA to the code users, but also between code users. If this concept is to be successful, contributions are needed from the code users. The next newsletter is scheduled for December 2006 and we would like to include a brief summary of your RETRAN and VIPRE activities. Please provide your contribution to CSA, P. O. Box 51596, Idaho Falls, ID 83405, or to one of the email addresses below by December 4, 2006. ***Contributors of a feature article will receive a RETRAN polo shirt.*** We are looking forward to hearing from all RETRAN and VIPRE licensees.

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Garry Gose	gcg@csai.com
Pam Richardson	pam@csai.com

The RETRAN web page is located at

http://www.csai.com/retran/index.html.

The VIPRE web page is located at

http://www.csai.com/vipre/index.html

Previous issues of the RETRAN/VIPRE Newsletter are available from the RETRAN or VIPRE web pages.

Steering Committee Members

Gregg Swindlehurst, Duke Energy (Chairman), gbswindl@duke-energy.com
Andres Gomez Navarro, Iberdrola, agn@iberinco.com
Sama Bilbao y Leon, Dominion, Sama_Bilbao@dom.com
David Huegel, Westinghouse, huegelds@westinghouse.com
Adi Irani, Entergy Nuclear Northeast, airani@entergy.com
Daren Chang, STPNOC, dchang@stpegs.com

Calendar of Events

User Group Meeting:
November 6-7, 2006
Richmond, Virginia
Details will be emailed to Maintenance Group Members