What Is VIPRE-01?
The VIPRE-01 code has the capability to perform very detailed nuclear reactor core thermal-hydraulic calculations to obtain minimum departure from nucleate boiling ratio (MDNBR) for steady-state conditions and operational transients. It compliments reactor system transient codes such as RETRAN-3D, which perform system transient calculations, but are unable to perform the detailed core channel and fuel pin calculations for MDNBR.

VIPRE-01 is a finite-volume sub-channel analysis code capable of three-dimensional modeling of reactor cores and other similar geometries in steady-state and transient conditions. VIPRE-01 calculates the detailed steady-state and operational transient core flow distributions, coolant conditions, fuel rod temperature and MDNBR. VIPRE-01 parentage is from the COBRA series of codes. The COBRA codes were not designed to meet the nuclear utilities needs and they were not adequately validated and documented. VIPRE-01 built on the COBRA code strength with upgraded models, numerics, documentation, and flexibility to meet the utilities’ analytical and software quality requirements.

Several U.S. utilities and international organizations use VIPRE-01 to evaluate thermal margins as reflected in the licensing submittals to the US Nuclear Regulatory Commission (USNRC), published papers, and VIPRE-01 User Group membership. Westinghouse has also adopted VIPRE-01 to perform MDNBR calculations for their customers.

In a partnership with sponsoring organizations, Electric Power Research Institute (EPRI) funded the VIPRE-01 project while modeling features and code development were directed by the sponsoring organization VIPRE users. EPRI sponsors have required carefully planned code development activities that have resulted in complete documentation, extensive validation and verification efforts, and carefully controlled maintenance procedures. Released versions of the VIPRE-01 code are maintained under a formal quality assurance (QA) program to ensure that modifications are properly made and that they do not yield unexpected changes to VIPRE-01 analysis results.

The VIPRE-01 code is operational on PCs and Linux workstations.

Why VIPRE-01?
VIPRE-01 is the only US system transient computer code whose development was funded and continues to be funded by commercial nuclear power plant owners. The Zachry Analysis Division - Idaho Falls office manages the VIPRE-01 User Group and is the commercialization agent for VIPRE-01. This ensures that the code features and capability were focused on the need of this commercial industry group to support plant operation, reload licensing and generic safety issues.

The US Nuclear Regulatory Commission performed extensive review VIPRE-01 and has issued a Safety Evaluation Report (SER) that finds it “acceptable for referencing in licensing applications”.

There are several intangible features that make the VIPRE-01 well suited for organizations looking for a well-documented and stable transient analysis tool. These are:

- **Established Track Record** – The VIPRE-01 code has been used by the nuclear industry for twenty years to support plant operation. Several organizations have obtained USNRC approval of Transient Topical Methodologies with VIPRE-01 that permits them to perform MDNBR calculations.
**Documentation** – The VIPRE-01 code is well documented. The documentation set includes five volumes:

Volume 1: Mathematical Modeling,
Volume 2: User's Manual,
Volume 3: Programmer's Manual,
Volume 4: Applications Manual, and
Volume 5: User Guidelines.

These manuals are kept under the same configuration control procedures as the VIPRE-01 code, so that the manuals are completely consistent with the current VIPRE version.

**Applicable to Multiple Plant Designs** – The VIPRE-01 code can be used for both pressurized water and boiling water reactor core transients. For organizations that own different nuclear facility designs, this will eliminate the need to learn different code packages.

**Well Organized Maintenance Group** – Members of the VIPRE-01 User Group have periodic User Group meetings at which presentations are given on analyses of interest, code development activities, and recent industry issues. These meetings provide a forum for VIPRE-01 code users to share information and discuss ideas. Information is distributed through published newsletters and available from the Zachry Idaho Falls web site (www.csai.com). Trouble report information is also available on this web site and is updated monthly. User Group Members have direct access to the code developers to help resolve modeling questions, modeling problems and suspected code errors.

**Configuration Control** – The VIPRE-01 code versions are carefully maintained and controlled. Periodic releases (as needed) of the code are made that contain model enhancements and error corrections. All changes are made under formal QA procedures. Each code release is extensively checked to ensure that results from analysis with prior versions are reproducible. This minimizes the chance of invalidating analysis results from earlier versions.

**VIPRE-01 Modeling Capability**

VIPRE-01 modeling structure is based on subchannel analysis. The VIPRE-01 model consists of the core or a section of core symmetry is simulated as an array of parallel flow channels with lateral connections between adjacent channels. A channel may represent a true sub-channel within a fuel rod array, a closed tube, or a larger flow area representing several sub-channels or fuel rod bundles. The analyst has a great deal of flexibility for modeling reactor cores or any other fluid flow geometry.

VIPRE-01 is limited to analysis of reactor cores. The core inlet fluid conditions must be obtained from system analysis codes like RETRAN-3D.

VIPRE-01 can compute single-phase and homogeneous two-phase flow from subcooled to superheated and supercritical conditions in water using the homogeneous equilibrium (three-equation) assumptions with empirical subcooled quality correlations for subcooled boiling and void-quality relationships to approximate the effects of the presence of two-phase. A wide variety of correlations is available for boiling heat transfer. Wall friction can be computed for fluids other than water through the code's capability to input a properties table.
A finite-volume conduction model is also used to compute temperature distributions and surface heat flux for walls, hollow tubes, cylindrical rods, and nuclear fuel rods. Internal UO\textsubscript{2} and Zircaloy thermal properties functions are installed and properties of other materials may be specified by input. For nuclear fuel rods, a dynamic fuel-clad gap heat conduction model is available to account for the effects of thermal expansion and internal pressure.

The core power is specified in terms of an average power, with radial power factors and axial power profiles for each rod.

Models added to the code for use specifically in BWR applications are (1) water tube channel modeling, (2) leakage flow path connection, and (3) a drift flux model.

The water tube channel model simulates the behavior of the hollow "water" rods that replace fuel rods in some BWR designs. The finite difference mass energy axial, and lateral momentum equations are modified to account for flow into and out of the water tube. The lateral leakage paths can be modeled in VIPRE as gap connections. The lateral leakage path requires the definition of a lateral momentum cell, with a specific width and area.

The general familiarity and acceptance of VIPRE-01 within the nuclear power industry, the formalized maintenance program, the existence of so many validated plant models and the flexibility of the code all are strong evidence that the code will remain a preferred analysis tool.